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Alternative Disposal Concepts for High-Level and Transuranic Radioactive Waste Disposal

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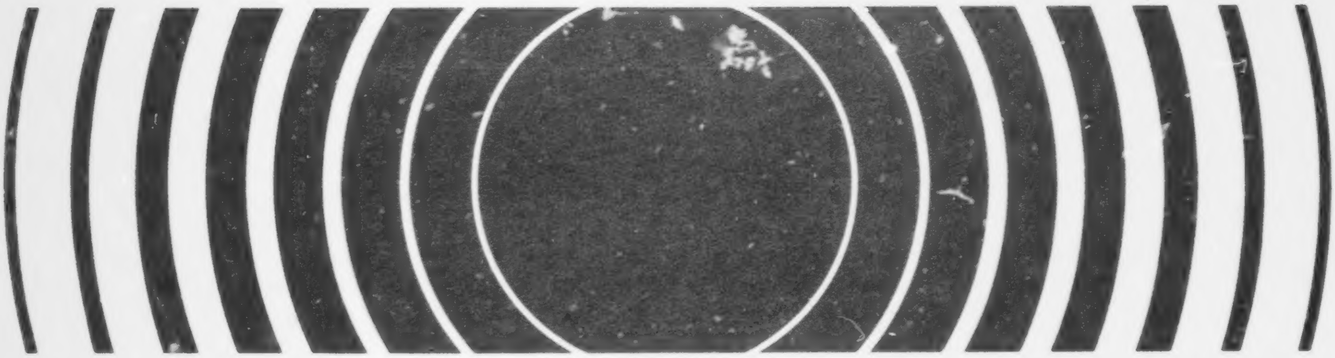
ALTERNATIVE DISPOSAL CONCEPTS FOR HIGH-LEVEL AND TRANSURANIC RADIOACTIVE WASTE DISPOSAL



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FOREWORD

The Office of Radiation Programs carries out a national program designed to evaluate the exposure of man to ionizing and nonionizing radiation, and to promote the development of controls necessary to protect the public health and safety, and to assure environmental quality.

As part of this program, the office is developing standards for the management and disposal of high-level radioactive wastes. A knowledge of available technologies and their capabilities is necessary for the development. This contract report examines a number of technologies which have been proposed as alternatives to disposal of high-level wastes in mined geological repositories.

Comments on this examination are welcomed; they may be sent to the Director, Criteria and Standards Division (ANR-460), Office of Radiation Programs, U.S. Environmental Protection Agency, Washington, D.C., 20460.

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ABSTRACT

Various alternatives have been proposed for the disposal of high-level and transuranic radioactive waste generated from the nuclear electric power industry and the U.S. Defense program. The most advanced disposal option, and the one under active development, is the U.S. owned and operated deep-mined geologic repository. This report reviews the primary alternative concepts to the geologic repository, their present state-of-development and, to the extent possible, their environmental implications. The concepts included are: transmutation, extraterrestrial disposal, seabed disposal, ice sheet disposal, and other continental geologic disposal (matrix of drilled holes, etc.). Projections of radioactive waste quantities and the technologies for partitioning and fractionation of the waste are also discussed.

This study reviewed information which was available through approximately January of 1978.

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1.0 INTRODUCTION

One of the major environmental, health, and safety concerns related to nuclear power is the permanent disposal of radioactive wastes. In particular, spent reactor fuel or reprocessed fuel waste is characterized by high levels of radioactivity, with some fission products and transuranic radionuclei remaining as hazardous substances for more than a million years. Because of the hazard to human health from radioactive wastes, these wastes must be placed in disposal sites capable of containment for periods approaching geologic time scales.

The Office of Radiation Programs of the U.S. Environmental Protection Agency (EPA) has a primary responsibility to establish radiation protection standards. In carrying out this responsibility, the EPA must assess the public health and the environmental impact of radiation from all sources in the United States.

This study supports EPA's assessment of radioactive waste for purposes of establishing environmental protection standards. It is one of several concurrent studies sponsored by EPA in the evaluation of high-level and transuranic waste. These companion studies include a MITRE study, Assessment of Waste Management of the Volatile Radionuclides[1] and the Arthur D. Little Inc. study, Technical Support for Radiation Standards for High-Level Radioactive Waste Management.[2]

The Arthur D. Little (ADL) study provides a technical assessment of the proposed U.S. disposal approach of placing high-level and transuranic radioactive waste in stable deep geologic formations. For purposes of this report, this concept is referred to as deep-mined geological repositories. The intent of the report is to examine alternative methods proposed for the disposal of high-level and transuranic radioactive waste. These alternative concepts include the following:

- Transmutation (Section 5) - nuclear conversion of radioisotopes to non-radioactive or short half-life isotopes
- Extraterrestrial Disposal (Section 6) - removal of waste from the earth and disposal in space or on planetary bodies
- Seabed Disposal (Section 7) - placement of the waste in the seabed thereby utilizing the ocean as an additional barrier between the waste and man
- Other Continental Disposal - alternative methods for disposal of waste on the earth land masses

For presentation purposes, the continental disposal is further separated into an ice sheet disposal concept (Section 8) and continental geological disposal concepts (Section 9). Because several disposal concepts require the separation of the waste into radionuclide groupings, Section 4 discusses the technology of partitioning and fractionation of radioactive waste. Section 3 provides background on the quantities and forms of radioactive waste for disposal. Section 2 provides a summary of the disposal concepts and compares the merits of the alternative approaches.

At present there is no accepted method for the final disposal of high-level or transuranic radioactive waste. The placement of these wastes in deep mines in geologically stable formations is the most technically developed concept and therefore the one which offers the most promise for early application. In this concept, a stable dry geological formation is to be selected and the radioactive waste emplaced in a mined area. These deep mined repositories can serve as interim storage areas until the long term isolation capability of the facility is confirmed or separate rooms may be backfilled as the waste is emplaced with eventual sealing of all openings for final disposal. The ADL study examines this concept in detail. The reader is referred to reference 2 for a discussion of the geological repository disposal concept. A health risk assessment for geological respository disposal is presently being prepared by EPA.

The difficulties in designating a final disposal method arise primarily from the need to assure that these highly radiotoxic wastes will be isolated from the biosphere for many thousands of years. Predictions of geological and hydrological behavior over such time periods are at best difficult and involve a large degree of uncertainty. Research and development must, however, provide reasonable assurance that the risk to present and future generations will be acceptable. The definition "acceptable" is an issue unto itself which has been and is being addressed by EPA.[3,4,5]

The form of the waste is significant in determining the method of final disposal. As originally conceived, the spent fuel elements are chemically processed to recover the usable uranium and plutonium as a part of the nuclear fuel cycle. A high-level radioactive residue results from this reprocessing operation. The residue, an aqueous raffinate, could be treated in several ways to form solids of different degrees of leach resistance and further packaged for final disposal. However, increasing concerns as to the potential for diversion of nuclear materials to weapons production has resulted in a moratorium on the reprocessing of commercial fuel in the U.S.[6] This decision produces substantial uncertainty as to the direction of nuclear waste management programs. Spent fuel elements may or may not be disposed of directly and reprocessed waste may or may not be available for further treatment to meet the requirements of various disposal options. Thus, the disposal of intact spent fuel must be considered as a possible requirement.

The proliferation issue and potential for diversion of nuclear materials are affecting the development and implementation of the U.S. nuclear program. Different fuel cycles and reactor types are presently under consideration. Development of the uranium-plutonium fuel cycle may not occur in favor of the establishment of a throwaway "cycle" (direct disposal of spent fuel elements), or the uranium-thorium cycle. These latter fuel cycles have advantages in limiting the accessibility to weapons grade material. Nuclear reactor types may continue with the Light Water Reactors (LWR) to simply utilize the uranium-235 resources, may shift to Heavy Water (D_2O) Reactors to obtain greater utilization or burnup of the fissile U-235, or may shift to the Gas Cooled Reactor (GCR) or Light Water Breeder Reactor (LWBR) but in centralized energy generating and fuel processing parks where safeguards are more easily implemented. For the different reactor types, different fuel handling and processing facilities will be required.

All of the above and more will affect the quantities, type, and form of the nuclear waste. In proceeding with a discussion of alternative nuclear waste management concepts, it must be borne in mind that there are many steps of research development and design. Final selection, evaluation, and implementation is a complex process and is influenced by economic, political, and technical factors.

Obviously, problems remain of both a technical and political nature that must be resolved to determine the most appropriate disposal method. For the present, therefore, it is prudent to continue to consider each of the possible radioactive waste disposal methods and to assume that processing of the spent fuel may be implemented either for purposes of fuel recycle or for preparation of the wastes for disposal.

Finally, although the studies presented herein are directed toward radioactive waste, the treatment and disposal of other toxic waste produced by man's activities are no less a concern. The concepts, the problems encountered, and solutions derived for radioactive waste will probably find application to treatment of waste from other sources.

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3. "U.S. EPA, Proceedings: A Workshop on Policy and Technical Issues Pertinent to the Development of Environmental Protection Criteria For Radioactive Wastes," Report: ORP/CSD-77-1, Reston, Va. (1977).

4. "U.S. EPA, Proceedings: A Workshop on Policy and Technical Issues Pertinent to the Development of Environmental Protection Criteria for Radioactive Waste," Report: ORP/CSD-77-2, Albuquerque, NM (1977).

5. U.S. EPA, Background Report: "Considerations of Environmental Protection Criteria for Radioactive Waste," February 1978.

6. Statement by President Carter on Nuclear Power Policy, The White House, April 7, 1977.

2.0 SUMMARY AND DISCUSSION

High-level and transuranic radioactive waste is created in the commercial nuclear industry and the U.S. defense programs. High-level waste in the context of this report is the highly radioactive liquid, containing fission products and actinides, which is the residue from the reprocessing to recover the uranium and plutonium from the spent fuel. High-level waste may also refer to the unreprocessed spent fuel elements in the throwaway "cycle." The bulk of this radioactive waste by the year 2000 will be from spent fuel discharged from nuclear electric generating plants. This waste is characterized by high specific radioactivity and is of particular concern to human health and the ecosystem since some fission products and produced radioactive actinide isotopes remain hazardous for hundreds to millions of years.

The most developed concept so far for the disposal of high-level and transuranic radioactive waste is the deep-mind geologic repository. This method has reached the facility design and site selection stage. Extensive technology research and development have been undertaken and studies of geologic formations have been and are being conducted to ensure the long-term isolation of waste from the biosphere. Many alternative approaches to the final disposal of high-level and transuranic radioactive waste have been proposed. While none of these alternatives is as advanced as the deep-mined geologic repository, they may supplement or replace this method at some future time if proven technically and economically practical and environmentally acceptable.

There are several alternative disposal concepts that have been considered in this report:

- Transmutation
- Extraterrestrial Disposal
- Seabed Disposal
- Ice Sheet Disposal
- Alternate Geologic Disposal Concepts

2.1 Disposal Options

2.1.1 Transmutation

Transmutation is the conversion of a radionuclide of undesirable characteristics (long life or high toxicity) to a different nuclear species by nuclear processes. The transmuted nuclide would have more favorable characteristics for disposal by forming a stable or short-lived isotope. Transuranic elements could be converted to a fissile isotope which could be fissioned or recycled.

Several methods are considered for transmutation of radionuclides:

- Particle accelerators
- Thermonuclear or fission explosives
- Fusion reactors
- Fission reactors

2.1.1.1 Particle Accelerators. Transmutation by particle accelerators, while feasible, has not been determined to be practical. The associated problems are high energy usage which can exceed the energy generated in producing the waste, expected high cost, and radioactive contamination.

2.1.1.2 Nuclear Explosives. It has been estimated that eleven one-hundred kiloton nuclear detonations per year would be required for transmutations of the long-lived fission products from each 1000 MWe reactor. It is not considered likely that this method of waste disposal would be considered acceptable.

2.1.1.3 Fusion Reactors. Fusion reactors potentially have very high neutron flux levels (10^{15} to 10^{16} neutrons/cm^2 -sec). The high energy neutrons produced in fusion reactors can be used directly to cause neutron induced reactions, or thermalized for capture in fission processes. Fluxes of this order of magnitude raise the possibility of transmuting not only actinides but also fission product nuclides such as Kr-85, Zr-93, Tc-99, and I-129, which are not considered practical for transmutation in fission reactors.

A sustained fusion, or thermonuclear, reaction has not yet been achieved. A major breakthrough is required before this technology can be realized.

2.1.1.4 Fission Reactors. Transmutation in fission reactors entails removal of the selected radionuclides from the spent fuel, fabrication into new fuel elements or separate elements, and irradiation to achieve transmutation by neutron capture.

Studies performed to date indicate that the transmutation of important actinides (Np, Am, Cm, Pu, Bk and Cf) is feasible. Research, development, and design studies are required, however, to implement this technology. In particular, reaction cross-section need to be measured, reprocessing and partitioning techniques have to be developed or perfected, and reactor designs must be developed and tested.

Transmutation in fission reactors of important long-lived fission products does not appear practical in that substantial removal of the radionuclides in reasonable time periods is not achievable. For example, reduction of the Tc-99 by a factor of 1000 could require 165 years, and to 10 percent 55 years. Specially designed reactors that obtain high thermal neutron fluxes from fast neutron reactors could conceivably reduce the irradiation time. The practicality of reactors of this type has not been evaluated.

2.1.2 Extraterrestrial Disposal

The concept of extraterrestrial disposal consists of placing a capsule containing the waste in space where further contact with earth is essentially eliminated. The space shuttle is being considered as the launch vehicle for extraterrestrial disposal because of its lower cost, and the added safety and reliability of a manned spacecraft.

Several disposal destinations have been studied:

- High earth orbits
- Solar orbits
- Solar system escape
- Solar impact
- Lunar impact or landing
- Planetary impacts

High earth orbits are unattractive since there is a possibility that the earth will recapture the waste. Solar orbits are possible for waste disposal, however, a portion of the solar system could become contaminated following failure of the waste capsule. Solar impact is not practical with currently available space vehicles. Planetary impacts are ruled out at the present time by international agreements. Solar system escape would provide complete disposal of the waste from the earth and solar system.

Lunar landings offer some advantage in that the waste could be stored with minimal risk and later recovered or launched to other space destinations. International agreements would be required for lunar disposal.

Extraterrestrial disposal of all high-level and transuranium waste to be generated in nuclear power reactors is not currently feasible. This would require an excessive number of launch operations, and both the cost and environmental effects would preclude such an operation. Environmental effects include noise and sonic booms, acidic rain, reduction in upper atmosphere ion concentrations and the local community interactions. The environmental effects are expected to be of minimal significance for the normal anticipated operations of some 50 to 100 flights per year by the year 2000 and the additional launches that might be conducted for selected waste disposal.

Extraterrestrial disposal of the actinides and separated long-lived fission products is considered feasible, but not necessarily practical. Studies presently being conducted by NASA are investigating space disposal concepts for the fission products and transuranic waste with uranium reclaimed. Depending on the composition of the waste, 100 to 250 space shuttle launches per year might be required by the year 2000. The number of launches is affected by the degree of separation of the fission products from the actinides, the age, and the method of encapsulation and radiation shielding of the waste.

The encapsulation and reentry shield must be designed for maximum containment of the waste even in the event of a catastrophic launch vehicle explosion and fire or reentry of the waste capsule into the earth's atmosphere. The additional weight required for this protection reduces the payload per launch resulting in increased cost and increased number of launches for extraterrestrial disposal.

The major disadvantages of extraterrestrial disposal are the potential for accidents and the cost. The waste form, the encapsulation method, the launch system, and the mission profile for extraterrestrial disposal have not been sufficiently defined for an analysis of accidents and their consequences. Preliminary worst case analyses do, however, indicate that they are potentially serious. Accident risk can be substantially reduced by system design and limitation on the quantities or types of waste for space disposal. Whether space disposal of waste with the necessary safeguards is economical, or whether the risk of accidents is acceptable, will require extensive study. The risk of accidents with the potential for releasing radioactive materials directly into the environmental must be carefully evaluated.

2.1.3 Seabed Disposal

Seabed disposal is the controlled emplacement of radioactive waste in deep sea sediments or rock formations under the ocean. The ocean floors are divided into three principal physiographic provinces: Continental Margin, Midoceanic Ridge, and Ocean Basin Floors. Some of these areas may contain possible locations for controlled emplacement of high-level radioactive waste. Potential sites for high-level waste disposal will be selected on the basis of high geologic stability; predictability for geologic time periods; limited resource potential; biological nonproductivity and sediment characteristics which are effective as barriers to radionuclide migration. Based on sediment data from numerous drilling experiments, seismic profiles, and bottom sediment photographs, the ocean areas in the middle of the tectonic plates and the middle of the ocean gyres (mid-plate/mid-gyre) exhibit characteristics which are particularly attractive as seabed disposal sites. Sediment sampling experiments are currently underway at two designated sites located in the middle of the North Pacific mid-gyre region to establish the suitability of these areas for high-level waste repositories.

An exact procedure for emplacement of radioactive waste canisters will not be chosen until seabed disposal has been determined to be feasible. However, several techniques are possible: free fall penetration in sediments, winch controlled emplacement in clay sediment, and drilled holes into underlying rock formations. Free fall penetration requires that the clay sediment have plastic properties which will collapse to fill the resultant cavity in reasonable time. Laboratory studies indicate that closure of the emplacement cavity would occur. In winch controlled emplacement, laboratory studies indicate that there may be some cavity closure problems and that a sealant may be required. Deep sea drilling from a surface ship has been demonstrated by several marine research centers. This emplacement technique has the advantage that many canisters could be placed in a single bedded area at depths of 100 to 500 meters. A hole sealant would be required. To date, drilling techniques using sealants for seabed disposal have not been demonstrated.

Much of the information needed to adequately assess the overall feasibility of seabed disposal is not available. There are several areas which require further information, particularly the ability of seabed disposal to act as a barrier to radionuclide migration:

- information on the characteristics of ocean provinces to determine and establish their overall suitability as potential seabed disposal sites
- technological capabilities including transportation, shipment, and placement of wastes
- leach rates for all radionuclides in proposed waste forms
- physical properties of deep sea sediments
- sorption and distribution coefficients of deep sea sediments
- retardation factors of sediments
- effects of thermal gradients on sediments (heat transfer properties)
- dynamic response of sediment to canister emplacement
- transport processes of radionuclides in deep sea sediments including structural and chemical properties and driving forces
- transport processes in the water column, including diffusion currents, advection, biological (feed web), and thermal plume

Because of the uncertainties associated with seabed disposal, it is not presently possible to conclude that this concept represents a practical long-term solution to the waste disposal problem.

2.1.4 Ice Sheet Disposal

Disposal of high-level and transuranic waste in the Antarctic and Greenland ice sheets has been proposed. The favorable features of ice sheet disposal are geographic isolation, relative isolation of the waste from inhabited areas in the event of waste leakage, low temperatures, and rapid heat dissipation.

Several waste emplacement concepts have been considered. A meltdown or free flow concept emplacement would be accomplished by predrilling a shallow hole and allowing the thermal heat of the waste canister to melt or free flow to the ice sheet basal. An anchored emplacement concept would provide for 200- to 500-meter-long cables anchored at the surface to hold the waste canister in place. A surface storage facility has also been considered. The surface storage facility would be mounted on jack-up pilings or piers resting on load bearing plates. Cooling of the canisters in a surface facility would be by natural air flow. Both the anchored emplacement and surface storage would provide for retrievability.

At the present time, there is insufficient knowledge of the physics and history of ice sheets. International groups of glaciologists concluded that ice sheets could not seriously be considered for radioactive waste disposal without further investigation in certain areas of limited knowledge:

- the evolutionary processes in ice sheets
- the relationships of ice sheet behavior with climatic changes
- the nature of future climatic changes on the stability of ice sheets

Ice sheets are not considered a feasible concept for the disposal of the long-lived radioactive waste at this time.

2.1.5 Continental Geologic Disposal

The continental geologic disposal concept is to place radioactive waste in stable geologic formations. The concept relies upon the long-term stability and the nuclide retention capability of the geology to isolate the waste for periods of millions of years. Since water is a primary transport mechanism, the selected geologic formations must be essentially free of groundwater movement.

The deep-mined geologic repository is the most advanced concept for the disposal of high-level and transuranic radioactive waste. This concept has proceeded to the stage of facility design, and efforts are underway to locate a politically and geologically acceptable site. Deep salt deposits have received the most attention as a suitable disposal media because of their demonstrated stability over very long time periods, their homogeneity, and their capability of plastic flow which would tend to seal cracks or fissures that may develop from mining operations or as a result of temperature gradients. Crystalline rock formations such as granite or basalt, shales, limestones, and certain clay beds are also being considered for disposal sites.

The deep-mined geologic repository would consist of surface facilities to receive and handle the waste, and mines 300 to 1500 meters deep in the selected rock formations. Capability to repackage the waste, if required, would be included at the surface facility.

The waste would be emplaced in the floor of the mine shafts at spacings limited by the heat production rate of the waste.

There are several alternative proposed geologic disposal concepts:

- solution mined cavities in salt deposits
- matrix of drilled holes
- super-deep holes
- deep well injection
- hydro-fracture
- rock melting concepts

Solution-mined cavities, matrix of drilled holes, and super-deep holes offer the possibility of deeper emplacement of waste than a repository which is limited by mine opening constraints. The random emplacement of waste packages in solution mining is such, however, that only low-heat rate waste such as actinides can be considered. The technology for super-deep boreholes has not been developed. A matrix of drilled holes requires the development of a hole sealant which will be an effective barrier to radionuclide transport.

Deep well injection and hydrofracture concepts may have application to low-level and intermediate-level liquid waste, but the long-term containment required for high level and transuranic waste has yet to be proven.

In rock-melting concepts, the heat of the radioactive waste melts the rock. The waste then descends to deeper depths or it mixes with the rock to form a waste-rock mix which eventually cools and solidifies. The rock-melting concepts require research and development to establish their practicality and to determine the physical characteristics and behavior of rock-waste mixes.

The problem of assessing isolation capability for alternative geologic disposal concepts is similar to those for the deep-mined geologic repository. Studies underway to determine the environmental acceptability of deep-mined geologic repository will therefore be of interest for other geologic disposal concepts.

The major areas of uncertainty in the deep-mined geologic repository are in the area of heat and mining effects on the host rock formation and the assurance that the radionuclides will not escape the repository as a result of natural events or accidental human intrusion over the long time periods required before they decay to innocuous levels.

2.2 Comparison of Disposal Concepts

There are several concepts for the disposal of high-level and transuranic radioactive waste which have the potential for eventual implementation:

- geologic disposal (primarily the deep-mined repository)
- seabed disposal
- extraterrestrial disposal for certain separated waste
- fission transmutation for actinides
- fusion transmutation for actinides and long-lived fission products

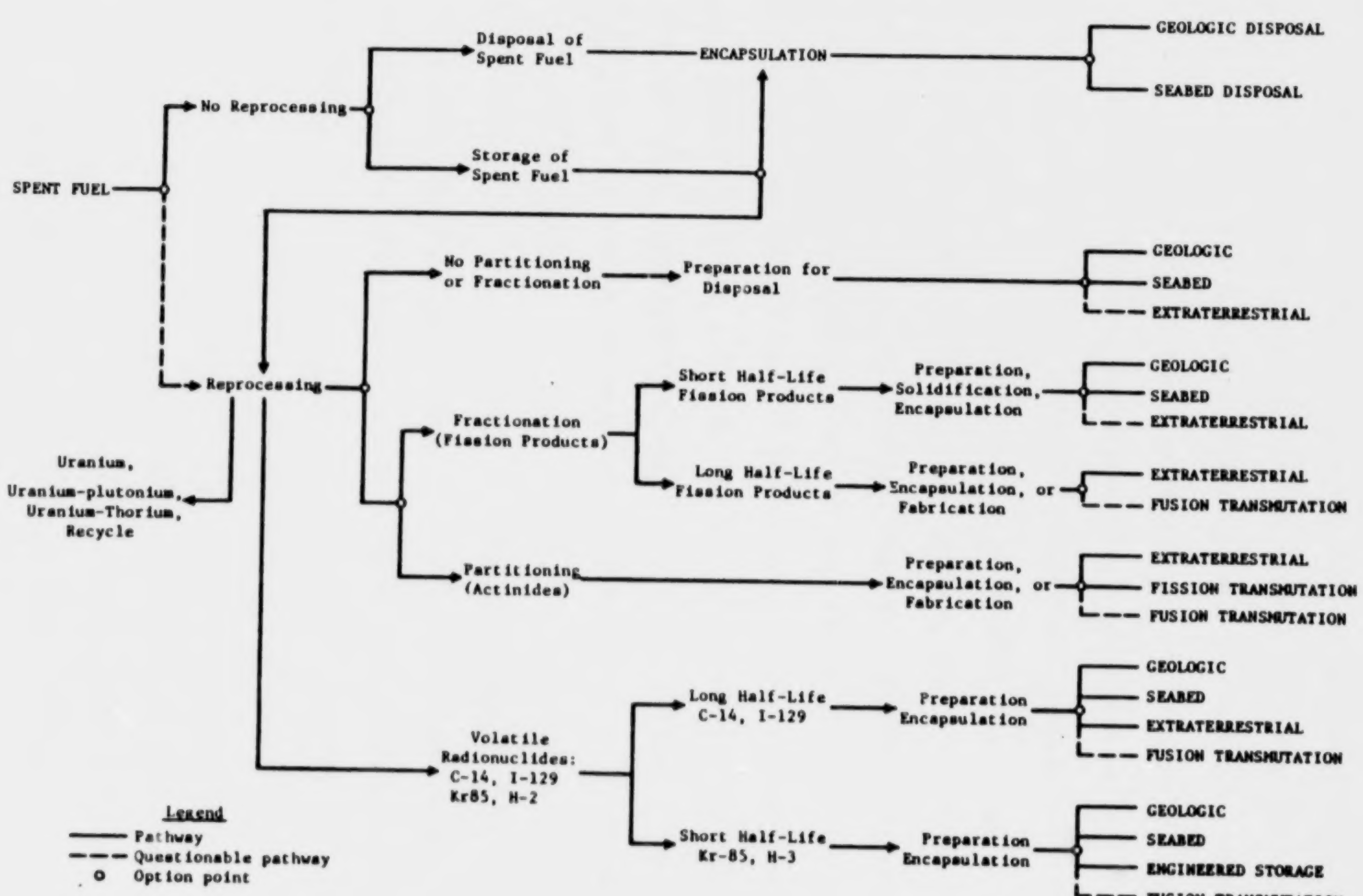
SPENT FUEL
No Reprocessing
Reprocessing
Disposal of Spent Fuel
Storage of Spent Fuel
ENCAPSULATION
GEOLOGIC DISPOSAL
SEABED DISPOSAL
Uranium, Uranium-plutonium, Uranium-Thorium, Recycle
No Partitioning or Fractionation
Preparation for Disposal
GEOLOGIC
SEABED
EXTRATERRESTRIAL
Fractionation (Fission Products)
Short Half-Life Fission Products
Preparation, Solidification, Encapsulation
GEOLOGIC
SEABED
EXTRATERRESTRIAL
Long Half-Life Fission Products
Preparation, Encapsulation, or Fabrication
EXTRATERRESTRIAL
FUSION TRANSMUTATION
Partitioning (Actinides)
Preparation, Encapsulation, or Fabrication
EXTRATERRESTRIAL
FISSION TRANSMUTATION
FUSION TRANSMUTATION
Volatile Radionuclides: C-14, I-129 Kr85, H-2
Long Half-Life C-14, I-129
Preparation Encapsulation
GEOLOGIC
SEABED
EXTRATERRESTRIAL
FUSION TRANSMUTATION
Short Half-Life Kr-85, H-3
Preparation Encapsulation
GEOLOGIC
SEABED
ENGINEERED STORAGE
FUSION TRANSMUTATION
Legend
Pathway
Questionable pathway
Option point

FIGURE 2-1
ALTERNATIVE WASTE DISPOSAL PATHWAYS

The various disposal pathways for commercial high-level and transuranic radioactive waste management are shown in Figure 2-1.

Present U.S. policy has deferred the reprocessing of commercial spent fuels. The spent fuel must, therefore, either be disposed of directly or committed to retrievable storage. Disposal options for spent fuel would be limited to geologic or seabed disposal. However, if the spent fuel is stored, it could eventually be returned for reprocessing and the alternative disposal options as indicated in Figure 2-1 would then be possible. The technical development and environmental studies of the alternative concepts have not advanced to a stage where quantitative comparisons can be made. In particular, the environmental, health, and safety aspects, as well as the probability for accidental release and the consequences of such releases, must be assessed for each step of the waste management process before a meaningful comparison can be made. The relative merits of the alternative disposal schemes are presented. No attempt is made herein to rank the desirability of the disposal options nor should any preference be implied.

The state of development, the major problems, and the advantages of the alternative disposal options are listed in Table 2-1.

The deep-mined geologic repository is the most advanced disposal concept and offers the earliest possibility for implementation. The major problem facing the acceptance of the deep-mined geologic repository is the reasonable demonstration that isolation can be maintained

TABLE 2-I

SUMMARY OF DISPOSAL CONCEPTS

Disposal Concept	State-of-Development	Major Problems	Advantages
Deep Geologic Disposal	Advanced state of development	Assurance of long term isolation is required	In an advanced stage of development
Alternate Geologic Disposal	Early stage of development	Proof of isolation, technical development is needed	Possible economics, deep disposal possible
Seabed Disposal	Early stage of development	Data is required for proof of concept and long term isolation	Added barrier to human environment and ocean dilution
Extraterrestrial Disposal	Early stage of development	High potential of accidents and accident consequences unknown	Elimination of long term uncertainty
Fission Reactor Transmutation	Early development for actinides, questionable application to fission products	Research, development and design needs	Elimination of long-lived actinides
Fusion Reactor Transmutation	Dependent on fusion reactor development	Major breakthrough in fusion development needed	Elimination of long-lived actinides and fission products

for thousands to millions of years. It must be determined that mining and the effects of waste heat will not result in pathways to the biosphere. Groundwater or radionuclide migration must be absent or of sufficiently slow rate that with the sorption capability of the host rock or other geologic media, radioactive materials are not transported to the environment in biologically significant quantities. Natural events, earthquakes, vulcanisms, meteorite impact, or accidental intrusions by man which would result in the release of the waste must be of negligibly low probability so as to be acceptable to society. Numerous studies are being conducted by the Environmental Protection Agency, the Department of Energy, the Nuclear Regulatory Commission, the Geological Survey and others to determine the acceptability of deep-mined geologic repositories for radioactive waste disposal. It is not intended nor is it within the scope of this study to evaluate the acceptability of deep-mined geological repositories.

Alternate continental geologic disposal concepts may have some advantages over repositories economically and perhaps in deeper emplacements of the waste. These options require technical development.

Seabed disposal is an attractive alternative in that an additional barrier exists between the waste and the human environment and few direct exposure pathways exist. For example, the oceans are not used for drinking water or for irrigation. The only direct exposure pathways are the ingestion of marine animals for food and some limited ingestion of marine plants. Ocean dilution of radionuclides which may escape the repository would also reduce the biological hazard. Data concerning the containment capabilities of seabed disposal is not presently adequate to implement this disposal method.

Extraterrestrial disposal removes the waste from the earth and with proper selection of space destinations essentially eliminates the uncertainty of future terrestrial contamination. There is, however, a potential for accidents in which the waste capsules may contaminate the earth. The probabilities (risk) and impacts (consequences) of accidents have not been analyzed. The risk and consequences can be minimized by design approaches although economics might be affected substantially, i.e., small amounts of actinides or long-lived fission products per launch. Further analysis is required before extraterrestrial disposal becomes an acceptable alternative.

Transmutation of the actinides in fission reactors has been considered an attractive disposal concept by researchers. The long-lived actinide radionuclides could essentially be eliminated by this approach. In the event that fusion reactors become practical, both the actinides and the long-lived fission products could conceivably be eliminated by transmutation utilizing the high neutron flux of these reactors. However, a major technical breakthrough is needed before fusion reactors can be considered practical and fission transmutation requires research, development, and design. Neutron absorption and reaction rates (cross-sections) with the long-lived actinides must be determined. Nuclear reactors must be developed, designed, and tested for the transmutation process. Further, partitioning, fractionation, and fabrication methods must be developed for the long-lived actinides and fission products.

In the extraterrestrial and transmutation process, it is likely that there will be short half-life waste that will require either geologic or seabed disposal. The time period for isolation will, however, be substantially reduced; from millions of years to perhaps a thousand years. The uncertainty of future events which might release the waste to the environment would correspondingly be reduced. While extraterrestrial disposal and transmutation appear as favorable concepts, it should be borne in mind that chemical separation and other processing facilities are not perfect in their operation. Some fraction of the long-lived radionuclides will remain with the shorter half-life material for terrestrial disposal. The added operating facilities will have some radioactive material releases and will increase the risk for accidents. These factors must also be considered in radioactive waste management and in the evaluation of disposal concepts.

2.3 Conclusions

Deep-mined geologic repositories offer the greatest potential as a near-term approach to final disposal of high-level and transuranic radioactive waste. In the event that repositories are deemed unacceptable for the final disposal of radioactive waste, or if disposal is deferred for other reasons, then it will be necessary to place the waste in long-term storage until alternative methods of disposal are developed and accepted. Storage of spent fuel elements is probably most desirable since it provides for the greater options of final disposal with the least economic penalty.

At present, none of the alternatives to geologic repositories have reached a stage of development to be considered acceptable methods of final disposal. They do, however, have potential for development to practical approaches and several would reduce the uncertainty of long-term containment.

Seabed disposal offers an additional barrier to transport of radioactive material to biologically active regions and provides dilution to reduce the biological hazard.

Extraterrestrial disposal and transmutation have the potential to remove the long-lived radionuclides from the earth and thus reduce the long-term uncertainty of waste disposal.

The possibility also exists of employing a multiple approach to radioactive waste disposal; a combiantion of fission transmuation of actinides, extraterrestrial disposal of selected long half-life fission products, and geologic or seabed disposal of short half-life radioactive waste is one example. Whether such an approach is economically, technically, or environmentally acceptable remains to be determined.

The discussion of disposal alternatives has been primarily directed to spent fuel from commercially operated reactors. Existing defense wastes are a special problem in that they exist in forms which are not readily adapted to further treatment. Extraterrestrial disposal and transmutation are therefore not likely to be attempted for these wastes. Accordingly, either geologic or seabed disposal would be anticipated for the Defense waste final disposal.

It has not been possible in this report to assess the radiological health risk of the alternative disposal concepts. Studies are presently being conducted by EPA, DOE, and others to determine the long-term health risk of geologic repositories. Similar studies are required for comparative evaluation of alternative disposal concepts.

3.0 QUANTITIES AND FORM OF HIGH-LEVEL AND TRANSURANIC WASTE

Radioactive waste may originate from a variety of sources: certain mineral processing activities; medical, industrial, and scientific radioisotope applications; nuclear power reactors; and U.S. Defense waste programs. This report deals with wastes from the spent fuel of nuclear power reactors and certain wastes from the U.S. Defense program. These wastes pose the greatest hazard to the environment and the long-term welfare of society. They are characterized by high specific radioactivity and contain elements of atomic number greater than 92 (transuranium elements). The transuranics are characterized by long half-life and high radiotoxicity and are therefore of particular concern.

At the present time, the Defense waste represents the greater bulk of waste for disposal. By the year 2000, the commercial waste will, however, far exceed the defense waste in total radioactivity for treatment and disposal, even if only the low projections of installed nuclear power are realized.

3.1 Present and Projected Quantitites of Waste

3.1.1 Existing Waste

The estimated inventory (in 1977) of spent fuel from operating commercial nuclear reactors is about 2,500 metric tons (MT).[1] This spent fuel is primarily stored at the reactor sites. In addition, there exist approximately 77 million gallons of Defense program high-level waste stored at government facilities at the Hanford

Reservation, Savannah River Reservation, and Idaho National Engineering Laboratory.[2] A backlog of 1800 MT of Defense program-related spent fuel has been accumulated from the Hanford N Reactor for processing and an additional amount of 400 to 900 MT/year is expected.[2] A small amount of high-level radioactive liquid waste is stored at the now shutdown Nuclear Fuel Services Plant at West Valley, New York.

The Defense waste, which represents the bulk of the present waste, exists in several different forms: solidified calcine powder (Idaho); salt cake; sludge; residual liquor (Hanford and Savannah River); and capsules of strontium and cesium (Hanford).

The quantities of fission products, actinides,* and contained sodium for each of the U.S. Government high-level waste storage sites are shown in Table 3-I.[2] The sodium is non-radioactive but is used in Defense waste programs in the form of NaOH to neutralize the nitric acid used in the treatment of irradiated fuel at Hanford and Savannah River. This permits the use of less expensive carbon steel tanks. The sodium is important, however, in that it complicates further processing for waste disposal as noted below.

The radionuclide content of the Defense waste is not well known (the program dates back to the 1940's) but representative compositions are given in Tables 3-II, 3-III, and 3-IV.

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*Actinides are elements of atomic number 89 or higher. They include the radioactive decay daughter products of the transuranium elements. Some of these isotopes and their daughter products are hazardous alpha radiation emitters.

TABLE 3-I

SUMMARY OF DEFENSE WASTE QUANTITIES

Site	Volumes Millions of Gallons	Radioactivity, Ci				Wt. (MT)			
		Sr-90 Plus Cs-137	Total FP's	Uranium	TRU	Total FP's	Uranium	TRU	Na Content
Hanford	51	2.4×10^8	2.5×10^8	7.1×10^2	1.4×10^5	60	900	.52	66,000
Savannah River	22	2.6×10^8	3.2×10^8	$\sim 8 \times 10^0$	7.4×10^5	57	$\sim$50	.44	30,000
Idaho	3	4.4×10^7	8.0×10^7	$\sim 1 \times 10^0$	1.0×10^3	9.2	$\sim$ 2	.02	30
TOTALS	76	5.4×10^8	6.5×10^8	7.1×10^2	8.8×10^5	130	952	.98	96,030

Source: Arthur D. Little, Inc., estimates, Reference 3

TABLE 3-II

INVENTORY OF MAJOR FISSION PRODUCTS AND ACTINIDES IN HANFORD HIGH-LEVEL WASTES DECAYED TO 1990

	Radioactivity (Ci)				
Radionuclide	**Salt Cake**	**Sludge**	**Residual Liquor**	**Capsules**	**Total**
Fission Products:					
H-3	*	*	1.1×10^4	-	1.1×10^4
C-14					$<1.6 \times 10^4$
Sr-90	2.0×10^6	4.5×10^7	6.0×10^5	5.8×10^7	1.06×10^8
Zr-93	*	6.9×10^3	*	-	6.9×10^3
Tc-99	*	*	3.1×10^4	-	3.1×10^4
Cd-113m	*	5.0×10^3	*	-	5.0×10^3
Sb-125	*	2.0×10^4	*	-	2.0×10^4
Sb-126	*	9.6×10^6	*	-	9.6×10^0
I-129	*	*	4.7×10^1	-	4.7×10^1
Cs-137	5.0×10^6	5.0×10^6	1.8×10^7	1.0×10^8	1.3×10^8
Ce-144	*	9.9×10^6	*	-	9.9×10^6
Pm-147	*	1.0×10^6	*	-	1.0×10^6
Sm-151	*	1.4×10^6	*	-	1.4×10^6
Eu-152	*	1.5×10^3	*	-	1.5×10^3
Eu-154	*	7.3×10^4	*	-	7.3×10^4
Eu-155	*	7.4×10^4	*	-	7.4×10^4
Actinides:					
U-233	*	4.0×10^2	*	-	4.0×10^2
U-235	*	1.3×10^1	*	-	1.3×10^1
U-236	*	3.0×10^2	*	-	3.0×10^2
Np-237	*	1.0×10^2	*	-	1.0×10^2
Pu-238		4.0×10^2	*	-	4.0×10^2
Pu-239		2.1×10^4	*	-	2.1×10^4
Pu-240		5.2×10^3	*	-	5.2×10^3
Pu-241		6.0×10^4	*	-	6.0×10^4
Am-241		5.0×10^4	*	-	5.0×10^4

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*Contains trace quantities of these isotopes.

Note: Daughter nuclides not listed; curie values are for parent nuclide only.

Source: ERDA-76-43, UC-70, "Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle," Volume 2, U.S. Energy Research and Development Administration, May 1976.

TABLE 3-III

RADIONUCLIDE CONTENT*
SAVANNAH RIVER HIGH-LEVEL WASTES (1985)

Radionuclide*	Total Activity (Ci)
Fission Products:	
Sr-90	1.3×10^8
Ru-106	1.8×10^6
Cs-137	1.3×10^8
Ce-144	1.1×10^7
Pm-147	4.6×10^7
Sm-151	4.2×10^6
Actinides:	
Pu-238	6.0×10^5
Pu-239	2.4×10^4
Am-241	6.0×10^4
Cm-244	6.0×10^4

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*Daughter nuclides in decay chains are not listed. Curie values are of important nuclides only.

Source: Alternatives for Long-Term Management of Defense High-Level Radioactive Waste--Savannah River Plant. ERDA 77-42/1, U.S. Energy Research and Development Administration, May 1977.

TABLE 3-IV

TYPICAL COMPOSITION OF CALCINED SOLIDS IDAHO CHEMICAL PROCESSING PLANT

	Composition, Wt. (%)	
	Aluminum (Non-fluoride) Waste	Zirconium (Fluoride) Waste
Al_2O_3	85	8
ZrO_2	0	34
HgO	1	0
B_2O_3	0.3	0.9
	2.4	0.1
Ca as CaF_2	0	54
Fission product and other oxides, fluorides	4.8	0.5
Nitrogen as N_2O_5	4	<1
H_2O	<2	<1
Bulk Density	1,100 kg/m^3	1,600 kg/m^3

Based on: Alternatives for Long-Term Management of High-Level Defense Waste--Idaho Chemical Processing Plant. Preliminary Draft, May 1977. Kearney, M.S., & Walton, R.D., Long Term Management of AEC/ERDA Generated High-Level Radioactive Waste, AIChE Symposium Series 154: 45-51, 1976. Reference 3.

3.1.2 Projected Quantities of Waste

The quantities of future waste are primarily dependent upon the growth of the commercial nuclear power industry. Estimates of installed nuclear power range from less than 400 to 1000 GWe in the year 2000. An estimate by S.M. Stoller Corporation is shown in Figure 3.1.[3] In 1976, the Energy Research and Development Agency (ERDA), now incorporated into the Department of Energy (DOE), projected nuclear generating capacity on a low growth scenario to 380 GWe in the year 2000.

Actual quantities of nuclear waste will be dependent upon a number of factors as previously noted; however, projections for a nominal case of 700 GWe to about the year 2010 are given in Figure 3.2. For reference purposes, light-water reactors typically discharge 25.5 MTH/GWe-year.* Included with this discharge is approximately 0.9 MT of fission products and 0.26 MT as transuranics (TRU). For the total lifetime (30 years) of 700 GWe added capacity, 5.36×10^6 MTHM of spent fuel would be discharged.[3]

The estimated range of total U.S. high-level waste to the year 2010 is shown in Table 3-V.[3]

The significant radionuclide composition of the commerical waste will vary with age. Table 3-VI presents the significant radionuclides (greater than 1 percent of total activity) for time periods up

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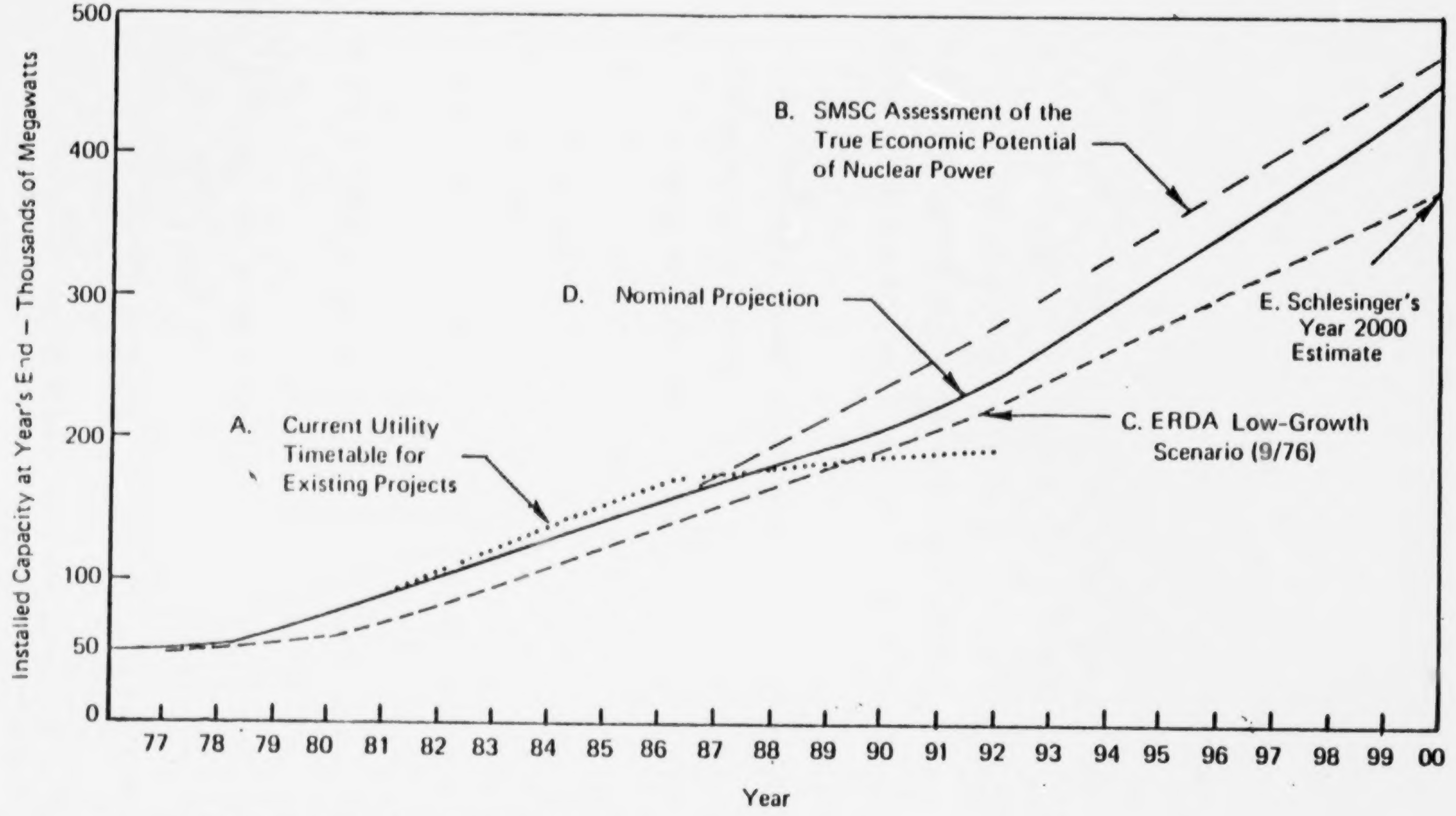
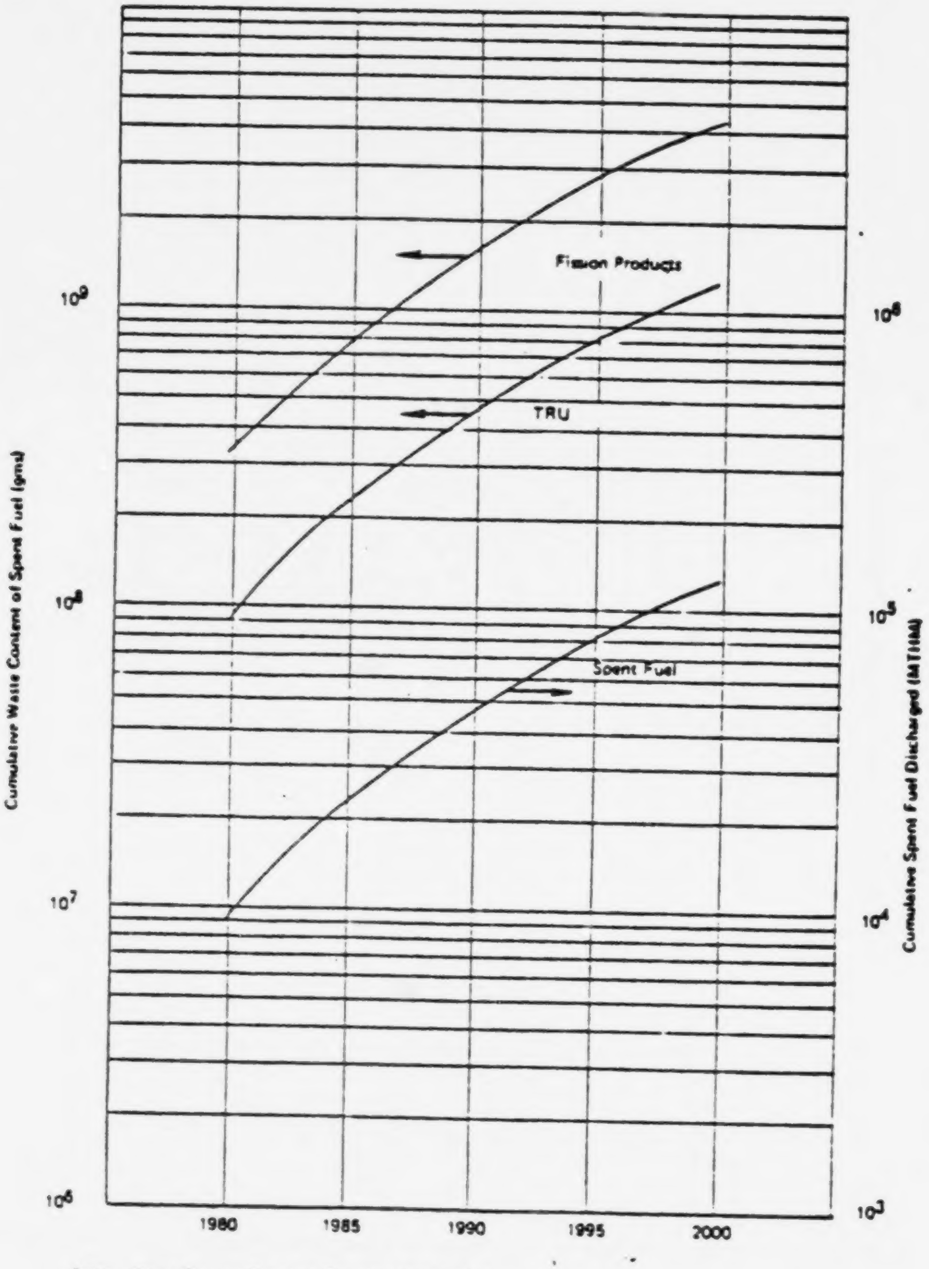
*1130 MWe PWR, 30-year lifetime, 70 percent capacity, 33 percent thermal efficiency. MTHM means metric tons (1000 kg) of Heavy Metal.



Source: The S. M. Stoller Corporation, 6/17/77.
Reference 3

FIGURE 3-1

U.S. NUCLEAR POWER GROWTH PROJECTION



Note. Waste Content Based on Ten-Year Decay Time.

FIGURE 3-2
PERSPECTIVE ON THE BUILDUP OF SPENT FUEL AND ASSOCIATED HIGH LEVEL WASTES VS. TIME (NOMINAL GROWTH CASE, THROWAWAY CYCLE)

Source: Arthur D. Little Inc., Reference 3

TABLE 3-V

PRINCIPAL RADIONUCLIDES IN WASTE--
THROWAWAY CYCLE

NUCLIDE	HALF-LIFE ①	RADIOACTIVITY AT VARIOUS DECAY TIMES, Ci/MTHM ②③					
		10(1) YEARS	10(2) YEARS	10(3) YEARS	10(4) YEARS	10(5) YEARS	10(6) YEARS
		HULLS					
H-3	12.26y	1.07(-1)	6.69(-4)	--	--	--	--
C-14	5730y	1.52(-2)	1.50(-2)	1.35(-2)	4.54(-3)	--	--
Fe-55	2.60y	1.69(2)	--	--	--	--	--
Co-60	5.26y	2.52(3)	--	--	--	--	--
Ni-59	8×10^4y	--	1.66(0)	--	1.52(0)	6.97(-1)	--
Ni-63	92y	2.36(2)	1.20(0)	1.36(-1)	--	--	--
Zr-93	1.5×10^6y	--	--	5.52(-2)	5.50(-2)	5.28(-2)	3.48(-2)
Nb-93m	13.6y	--	--	6.09(-2)	5.79(-2)	5.28(-2)	3.48(-2)
All Others		--	--	0.03(0)	0.02(0)	0.03(-1)	0.07(-2)
TOTAL		2.93(3)	1.22(2)	1.92(0)	1.65(0)	8.06(-1)	7.03(-2)
		FISSION PRODUCTS					
H-3	12.26y	4.16(2)	2.61(0)	--	--	--	--
Kr-85	10.76y	5.98(3)	--	--	--	--	--
Sr-90	27.7y	6.00(4)	6.52(3)	--	--	--	--
Y-90	64.0h	6.00(4)	6.52(3)	--	--	--	--
Zr-93	1.5×10^6y	--	--	1.86(0)	1.86(0)	1.78(0)	1.18(0)
Nb-93m	13.6y	--	--	1.36(0)	1.86(0)	1.78(0)	1.18(0)
Tc-99	2.12×10^5y	--	--	1.43(1)	1.38(1)	1.03(1)	5.44(-1)

TABLE 3-V (Cont.)

PRINCIPAL RADIONUCLIDES IN WASTE--

THROWAWAY CYCLE

NUCLIDE	HALF-LIFE ①	RADIOACTIVITY AT VARIOUS DECAY TIMES, Ci/MTHM ② ③					
		10(1) YEARS	10(2) YEARS	10(3) YEARS	10(4) YEARS	10(5) YEARS	10(6) YEARS
		FISSION PRODUCTS (concluded)					
Tc-99	2.12×10^5y	--	--	1.43(1)	1.38(1)	1.03(1)	5.44(-1)
Pd-107	7×10^6y	--	--	--	--	--	1.05(-1)
Sn-126	50m	--	--	5.60(-1)	5.26(-1)	2.82(-1)	--
Sb-126		--	--	5.60(-1)	5.26(-1)	2.82(-1)	--
Sb-126m		--	--	5.55(-1)	5.21(-1)	2.79(-1)	--
I-129	1.7×10^7y	--	--	--	--	--	3.62(-2)
Cs-134	2.046y	9.18(3)	--	--	--	--	--
Cs-135	3.0×10^6y	--	--	2.23(-1)	2.23(-1)	2.18(-1)	1.77(-1)
Cs-137	30.0y	8.64(4)	1.08(4)	--	--	--	--
Ba-137m	2.554m	8.08(4)	1.01(4)	--	--	--	--
Pm-147	2.62y	7.87(3)	--	--	--	--	--
Sm-151	87y	--	5.68(2)	4.37(-1)	--	--	--
Eu-154	16y	--	--	--	--	--	--
All Others		0.10(5)	0.01(4)	0.06(1)	0.07(1)	0.02(1)	--
TOTAL		3.20(5)	3.46(4)	2.09(1)	1.99(1)	1.52(1)	3.21(0)
		ACTINIDES AND DAUGHTERS					
Pb-209	3.30h	--	--	--	--	4.19(-1)	9.40(-1)
Pb-210	20.46	--	--	--	--	9.80(-1)	4.70(-1)
Pb-214	26.8m	--	--	--	--	9.80(-1)	4.70(-1)
Bi-210	5.013d	--	--	--	--	9.80(-1)	4.70(-1)
Bi-213	47m	--	--	--	--	4.19(-1)	9.40(-1)

TABLE 3-V (Cont.)
PRINCIPAL RADIONUCLIDES IN WASTE--
THROWAWAY CYCLE

NUCLIDE	HALF-LIFE ①	RADIOACTIVITY AT VARIOUS DECAY TIMES, Ci/MTHM ② ③					
		10(1) YEARS	10(2) YEARS	10(3) YEARS	10(4) YEARS	10(5) YEARS	10(6) YEARS
		ACTINIDES AND DAUGHTERS (continued)					
Bi-214	19.7m	--	--	--	--	9.80(-1)	4.70(-1)
Po-210	138.40d	--	--	--	--	9.80(-1)	4.70(-1)
Po-213	4.2 x 10^{-6}s	--	--	--	--	4.10(-1)	9.20(-1)
Po-214	1.64 x 10^{-4}s	--	--	--	--	9.80(-1)	4.70(-1)
Po-218	3.05m	--	--	--	--	9.80(-1)	4.70(-1)
At-217	3.23 x 10^{-2}s	--	--	--	--	4.19(-1)	9.40(-1)
Rn-222	3.8229d	--	--	--	--	9.80(-1)	4.70(-1)
Fr-221	4.8m	--	--	--	--	4.19(-1)	9.40(-1)
Ra-225	14.8d	--	--	--	--	4.19(-1)	9.40(-1)
Ra-226	1602y	--	--	--	--	9.80(-1)	4.70(-1)
Ac-225	10.0d	--	--	--	--	4.19(-1)	9.40(-1)
Th-229	7340y	--	--	--	--	4.19(-1)	9.40(-1)
Th-230	8.0 x 10^{4}y	--	--	--	--	9.78(-1)	4.70(-1)
Th-234	24.10d	--	--	--	--	--	3.14(-1)
Pa-233	27.0d	--	--	--	--	1.19(0)	8.86(-1)
Pa-234m	1.0175m	--	--	--	--	--	3.14(-1)
U-233	1.62 x 10^{5}y	--	--	--	--	4.18(-1)	9.40(-1)
U-234	2.47 x 10^{5}y	--	--	--	--	1.53(0)	4.11(-1)
U-236	2.39 x 10^{7}y	--	--	--	--	--	3.88(-1)
U-238	4.51 x 10^{9}y	--	--	--	--	--	3.14(-1)
Np-237	2.14 x 10^{6}y	--	--	--	--	1.19(0)	8.86(-1)
Np-239	2.346d	--	--	--	6.94(0)	--	--
Pu-238	86.4y	2.19(3)	1.09(3)	--	--	--	--
Pu-239	24,390y	--	3.30(2)	3.22(2)	2.52(2)	1.98(1)	--
Pu-240	6580y	--	4.87(2)	4.44(2)	1.77(2)	--	--

TABLE 3-V (Concluded)

PRINCIPAL RADIONUCLIDES IN WASTE--

THROAWAY CYCLE

NUCLIDE	HALF-LIFE ①	RADIOACTIVITY AT VARIOUS DECAY TIMES, Ci/MTHM ②③					
		10(1) YEARS	10(2) YEARS	10(3) YEARS	10(4) YEARS	10(5) YEARS	10(6) YEARS
		ACTINIDES AND DAUGHTERS (concluded)					
Pu-241	13.26y	7.95(4)	1.11(3)	--	--	--	--
Pu-242	3.79×10^5y	--	--	--	--	1.45(0)	2.80(-1)
Am-241	458y	1.73(3)	3.86(3)	9.24(2)	6.94(0)	--	--
Am-243	7.95×10^3y	--	--	--	--	--	--
Cm-244	17.6y	1.35(3)	--	--	--	--	--
All Others		0.08(4)	0.08(3)	0.05(3)	0.08(2)	0.15(1)	0.04(1)
TOTAL		8.56(4)	6.96(3)	1.73(3)	4.51(2)	4.03(1)	1.73(1)

1. Half-lives are reported in seconds (s), minutes (m), hours (h), days (d), and years (y).

2. Numbers in parentheses represent powers of ten.

3. Dashes indicate a value less than one percent of the total in a given column. Tritium and carbon-14 values are exceptions.

TABLE 3-VI

ESTIMATED RANGE OF TOTAL DOMESTIC HIGH-LEVEL WASTE BURDEN*
(CIRCA 2010)

	Category of Wastes	High Level Wastes				Other Associated Wastes		
		Spent Fuel (MTHM)	Total Radioactivity (Ci)	Fission Products (MT)	TRU (MT)	Iodine-129 (Ci)	Carbon-14 (Ci)	Miscellaneous[+] (Ci)
1a.	Commercial Wastes (throwaway fuel cycle)	$3.1\text{-}7.7 \times 10^5$	$1.3\text{-}3.2 \times 10^{11}$	11,000-27,000	3,100-7,700	(Contained in spent fuel)		
1b.	Commercial Wastes (mixed oxide recycle)	$3.1\text{-}7.7 \times 10^5$	$1.0\text{-}2.5 \times 10^{11}$	9,000-22,000	700-1,600	$1.3\text{-}3.2 \times 10^4$	$1.4\text{-}3.5 \times 10^5$	$0.9\text{-}2.3 \times 10^9$
2.	Waste from Defense Programs	-	6.5×10^8	130	1.2	n.a.	n.a.	n.a.

---

*Quantities of commercial wastes based on lifetime production for range of gross nuclear capacity additions (400-1,000 GW) keyed to LWR generation. Data are for 10-year-old wastes. Quantities and characteristics of non-commercial wastes keyed to existing inventory.

[+] "Miscellaneous" consists of: Cladding hulls, fuel assembly structure, entrapped TRU, and entrapped fission products.

Source: Arthur D. Little, Inc., Reference 3 to one million years for a throwaway fuel cycle. The fission products are the major source of radioactivity up to 100-200 years. Beyond 1000 years, among the fission products, only Zr-93, Tc-99, Pd-107, I-129, and Cs-135 are significant. The neutron activation radionuclides C-14 and Ni-59 also remain significant after 1000 years' decay. Short half-life daughter products from the radioactive decay of the long half-life processes are also significant contributors to the radioactivity beyond 1000 years.

Plutonium and americium are the primary radioactivity sources from about 200 to beyond 10,000 years. Past 100,000 years, the actinides daughter products become significant contributors to the radioactivity source. The alpha-emitting actinides are, of course, a potential major health hazard throughout their lifetimes.

3.2 Form of the Waste for Disposal

The form of the waste for disposal is dependent upon the policy decision regarding reprocessing and the disposal option ultimatel selected. The waste form can, however, be generically considered to be one of three types: 1) spent fuel; 2) solidified and packaged residue from reprocessing; 3) solidified and packaged-partitioned and fractionated waste.

3.2.1 Spent Fuel

Spent fuel may be treated in several ways in preparation for disposal. The fuel elements, following a period of aging to facilitate handling and to reduce the radioactivity and heat generation, would be encapsulated. It is also probable that portions of the fuel assemblies, i.e., nozzles, end boxes, etc., would be separated from the remaining hardware to reduce the total mass and volume. The fuel assembly hardware is initially contaminated with fission products and transuranium elements. A proposed standard requiring materials contaminated with greater than 10nCi/gm to be disposed of in a Federal repository may result in this material requiring the same disposal as solidified high-level waste, unless advanced methods of decontamination and transuranic element removal are developed.

The fuel elements could be melted and recast in a form which facilitates handling and disposal. This later option could be particularly important for disposal options such as seabed disposal where a specially formed waste capsule may be required. In the case of melting the fuel elements, consideration must also be given to the collection and disposal of volatile compounds that will be released. Relatively long-lived volatile radionuclides such as I-129, C-14, Kr-85, and tritium could be released.

3.2.2 Reprocessed Waste

Where reprocessing is performed to recover uranium, uranium and plutonium, or uranium-plutonium-thorium, the aqueous raffinate will be further treated to form a solid waste. Advanced forms of solidified waste are granularized calcine and glass.

The calcined product is of approximately the same volume as the liquid waste. The vitrification of waste requires the addition of borosilicate or phosphate glass. The waste glass form produced from high-level liquid waste is from 60-80 liters/MTU. The cladding and hulls of the fuel elements are treated and disposed of separately. If assumed to be compacted to 70 percent of the theoretical density, about 60 liters/MTU would be formed.[3]

The form of the Defense waste for final disposal has not as yet been specified. The INEL waste is at present a solid calcine and could readily be converted to the higher leach resistant glass form. The Hanford and Savannah River waste, however, has a high sodium content which makes the conversion to glass more difficult. In order to keep the Na content of the glass below 10 percent to facilitate conversion to glass, 10^6 metric tons of glass waste would be produced. If the Na is removed, the limiting factor is the uranium. At 40 percent, uranium plus fission product content, 2.8×10^3 metric tons would be produced. If the Na and uranium are removed, the waste for disposal would be only 300 metric tons of glass.[3]

The calcine requires packaging to contain the loose granules, however, all waste forms require containment to protect against exposure of workers and to provide radiation shielding during handling and shipping. The containment is also necessary to avoid leakage or contamination in the event of accidents and to provide resistance against corrosion and leaching of the waste in the disposal environment. Carbon steel, stainless steel, and titanium have been suggested as waste form encapsulation materials. Titanium has been projected to have the longest containment lifetime--up to 1,000 years.[3] The type of encapsulation material will be dependent upon the length of time that container integrity is determined to be required. Containment for several hundreds to 1000 years is adequate for isolation required for the shorter half-life fission products. However, the encapsulation material cannot assure containment for the long-lived fission products and transuranium elements. The packaging and encapsulation material is important in assuring containment for the period of time during which the waste may have to be retrieved.

In the reprocessing of spent fuel, certain volatile radionuclides will be released for which collection and immobilization technologies are under development. The volatile radionuclides of concern are: Kr-85, C-14, I-129, and tritium. The Kr-85 and tritium have relatively short half-lives and therefore require isolation from the environment for shorter periods of time--on the order of a hundred years. Possible forms for disposal of these radionuclides are listed below:

Radionuclide	Half-Life	Possible Disposal Form
Tritium (H-3)	12.26 y	Polymer impregnated concrete or Polyethylene organic compounds
Carbon-14	5730 y	$CaCO_3$ in concrete
Krypton-85	10.76 y	Carbon steel pressure vessels or Zeolite crystal lattice
Iodine-129	1.7×10^7 y	Barium Iodate incorporated in concrete

The disposal of these volatile radionuclides is discussed in reference 4.

3.2.3 Partitioned and Fractionated Waste

There are advantages to partitioning and fractionating the waste to separate the long half-life from the short half-life radionuclides. These separate fractions could possibly be disposed of by more economical methods. Partitioning and fractionation are required for the transmutation and extraterrestrial disposal methods.

For the transmutation disposal method, long-lived elements would be fabricated into targets for particle acceleration and fusion reactors or into fuel elements for exposure in fission reactors. Partitioned waste for extraterrestrial disposal would be encapsulated in special containers acceptable for space disposal (see Section 6). It is assumed that the residual material would be solidified in the calcine or glass form as noted above for reprocessed waste.

REFERENCES

1. Dr. T. English et. al., "An Analysis of the Technical Status of High Level Radioactive Waste and Spent Fuel Management Systems," JPL 77-69, Jet Propulsion Laboratory, Pasadena, Cal., December 1977.

2. "Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle," ERDA-76-43 Battelle, Pacific Northwest Laboratories, May 1976.

3. "Technical Support for the Radiation Standards for High-Level Radioactive Waste Management," Task A and B, Draft, Arthur D. Little, Inc.

4. P.M. Altomare et al., "Assessment of Waste Management of Volatile Radionuclides," MTR-7718, MITRE Corporation, McLean, Va., May 1979.

4.0 PARTITIONING AND FRACTIONATION

Radionuclides produced in nuclear power reactors include actinides (caused by neutron capture in the fertile materials) and fission and activation products. Their half-lives vary over a wide range--from minutes to millions of years. Current plans are to treat high-level waste (spent fuel elements or solidified reprocessing waste) as a single entity in storage, solidification, and disposal (temporary or permanent). This procedure may be adequate for disposing high-level wastes, but other disposal alternatives exist which require the waste to be separated into its components--actinides, fission products, and volatiles. The optimum waste system management could consist of several of the disposal alternatives discussed in this report. If the waste could be separated into fractions which have comparable half-lives, short-lived fractions might then be placed in deep-mined geological repositories where they would decay to innocuous levels in times during which isolation could more reasonably be assured, i.e., thousands of years. Long-lived fractions could be considered for other treatment: transmutation to short-lived, nonradioactive nuclides or fissile species; extraterrestrial; or other types of disposal. Initial considerations were based on the concept of minimizing the long-lived impurity content of short-lived fractions so that after a period of about a thousand years, the short-lived fraction would represent no significant radiological toxicity. The actual percentage of long-lived nuclides allowable would be determined by technical limitations and the particular nuclide, since not all are equally hazardous.

In separating the long-lived nuclides from the short-lived ones, emphasis has primarily been placed on separating the actinide elements from the fission products since these elements not only have long life-times but are also highly radiotoxic. The chemical separation of actinides from fission products is generally referred to as partitioning. In some cases it may be necessary to separate each individual type of nuclide both chemically and isotopically. For example, it is often desirable to separate the element curium from other actinides because of its intense radioactivity. The separation of individual elements from mixtures is referred to as fractionation. Partitioning and fractionation are appropriate only for reprocessed waste. Isotopic separation may be necessary in situations where the transmutation of a stable or relatively harmless isotope of a given chemical element tends to augment rather than reduce the radiological hazard. However, it must be noted that isotopic separation is an extremely expensive process by currently available techniques.

4.1 Chemical Processes

4.1.1 Spent Fuel Reprocessing

The radiological and chemical releases in partitioning are related to the chemical processes that are involved. In most cases the irradiated fuel is first dissolved in HNO_3 and the solution is fed to a solvent extraction stage where the Pu and U are separated from the other constituents and subsequently recovered. In most reprocessing plants, the primary extraction is done by the Purex process using tributyl phosphate (TBP) as the solvent. The residual waste solution from solvent extraction contains about 99.9 percent of the nonvolatile fission products and almost the whole original actinide content except U, Pu and some Np. The fraction of U and Pu reaching the waste stream depends on the efficiency of the separation process. A value between 0.1 and 0.5 percent is considered as a design objective by present methods, although present recoveries may be less. In addition to these, there are other chemical impurities such as organic solvents, nitric acid, and corrosion products from plant vessels.

The waste is treated to remove organic solvents and then concentrated by evaporation, because there is strong economic incentive to reduce the volume. The segregation of highly active wastes from low-level wastes and the minimizing of salts in the waste stream are of particular importance in volume reduction. Highly irradiated fuel from LWRs produces several hundred litres of waste/MT of fuel processed.

The nature of the hazard from the fission product differs from that due to actinide components because the actinides are, in general, alpha emitters, and are a primary health hazard only if ingested into the body. The fission products present both internal and external hazards. The alpha activity is initially dominated by curium isotopes, after several years decay, americium, and after several thousands of years, plutonium become the controlling actinide in terms of the number of curies. In the very long term (millions of years), Np-237 and U-238 have the greatest dose impacts.

In actinide partitioning, the main problem is the removal of Pu, Am, and Cm. It is necessary to maintain plutonium in an extractable form at very low concentrations because of its very high radio and chemical toxicity and to avoid the possibility of a criticality accident. With Am-Cm processing, the major difficulty is separating these elements without generating large amounts of radioactive chemical wastes. Experience in regard to the operation of radio-chemical plants which utilize extensive recycle of the waste streams is limited. Although the optimum process for each actinide has not yet been definitively established, removal of Cm, Np, and most of the plutonium by adding an extra extraction cycle to the Purex process is considered a strong possibility.

A multifaceted waste management scheme would require separation or partitioning of the high-level waste into its principle components--actinides and fission products. For a successful utilization of the disposal of radioactive wastes by the transmutation technique, such separation is an absolute necessity. The reason for this is the different neutronic behavior of actinides and fission products.

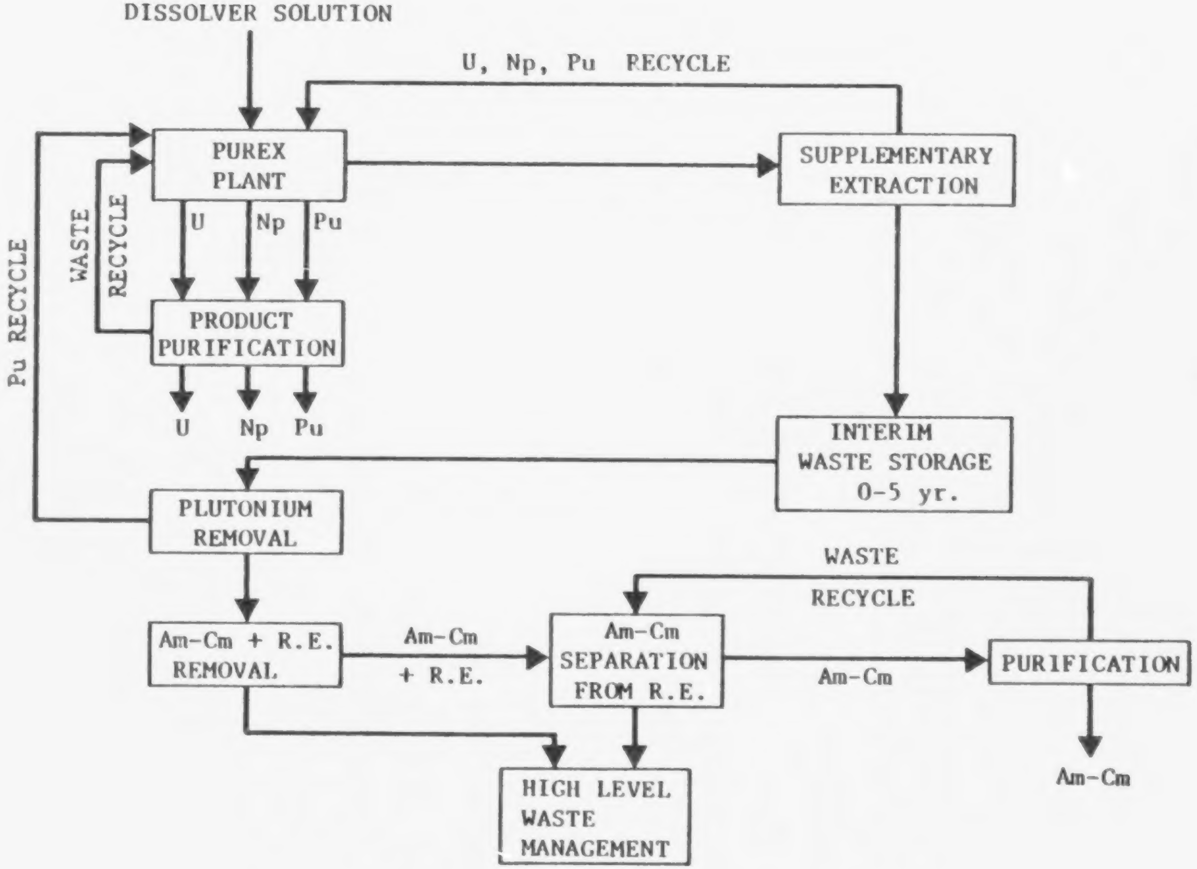
About 99.5 percent of the uranium and plutonium in the spent fuel of light-water reactors is recovered by present reprocessing techniques. The other 0.5 percent is lost to the high-level wastes. Studies on the feasibility of partitioning actinides from high-level wastes have been carried out at Battelle Northwest Laboratories, Oak Ridge National Laboratory, and EURATOM, in Ispra, Italy. Some specialized techniques are being developed at other laboratories, but the process developments have not progressed to the stage where it is possible to determine cost-benefit tradeoffs. The separation processes with the greatest potential are solvent extraction, ion exchange, and precipitation (or some combination of these methods).

4.1.2 Solvent Extraction

Solvent extraction is the most widely used technique because of a high degree of selectivity and purity of solvents available, the possibility of continuous operation, and the availability of a wide variety of suitable industrial scale extraction equipment with possibility of automation, remote control, high level of productivity, use of a wide range of concentrations, etc. There are, however, disadvantages such as the inflammability and toxicity of extraction liquids and the possibility of radiation damage to them, thus reducing their effectiveness.

4.1.2.1 Actinides

Figure 4-1 from a paper by Bond and Leuze[1], shows a conceptual processing sequence for actinide partitioning based on a combination of modified Purex processing and secondary processing of the high-level waste.


DISSOLVER SOLUTION
U, Np, Pu RECYCLE
PUREX PLANT
SUPPLEMENTARY EXTRACTION
WASTE RECYCLE
Pu RECYCLE
U
Np
Pu
PRODUCT PURIFICATION
U
Np
Pu
INTERIM WASTE STORAGE 0-5 yr.
PLUTONIUM REMOVAL
WASTE
RECYCLE
Am-Cm + R.E. REMOVAL
Am-Cm
+ R.E.
Am-Cm SEPARATION FROM R.E.
Am-Cm
PURIFICATION
Am-Cm
HIGH LEVEL WASTE MANAGEMENT

*R.E. - Rare Earths

Source: Bond and Leuze, Reference 1.

FIGURE 4-1

CONCEPTUAL PROCESSING SEQUENCE FOR ACTINIDE PARTITIONING

There are major radiological and chemical problems yet to be resolved:

- Recycle of low- and intermediate-level wastes in Purex
- Adequate U and Np recovery by Purex
- Recovery of actinides sorbed on solids and of "inextractable Pu"
- Adequate Am-Cm removal from waste without greatly increasing the waste volume
- Actinide recovery from miscellaneous wastes, burnable waste, cladding hulls, spent ion-exchange resin, HEPA filters, etc.

Careful process control will be necessary to ensure that actinides are not released along with fission products. For example, when Purex feed is stored at high temperature, zirconium and molybdenum salt crystals are formed which contain up to two percent plutonium. Also, zirconium hydrolyzes at high temperatures to form colloids that carry plutonium.

As yet, no simple solvent extraction method has been developed for partitioning all of the actinides. A multi-step solvent extraction process based on more than one solvent has the greatest possible chance of success. The processes for the extraction of U, Np, and Pu are different from that for Am and Cm.

The Purex process using tributyl phosphate (TBP) has been demonstrated on a plant scale for the separation of the U, Np, and Pu with recoveries of up to 99.9 percent, 90-95 percent, and 99.8 percent, respectively.[2]

Modifying the Purex process for complete separation of Am-Cm and higher transuranium elements from all fission products does not appear feasible, but separation above 95 percent of the trivalent (Am-Cm) actinides and lanthanides by TBP extraction from solutions heavily salted with metal nitrates has been achieved.[3]

Solvents with the greatest potential for the partitioning of Am and Cm are di-ethyl hexyl phosphoric acid, bidentate organophosphorus compounds, and dibutyl phosphonate. Solvent extraction using bidentate organophosphorous reagents for the removal of trivalent actinides and lanthanides from high-level purex waste is being experimented on at the Idaho Nuclear Engineering Laboratory.[4]

In solvent extraction, solvent additives to improve the degree of separation will often give rise to excessive amounts of inert materials harmful to waste processing or disposal. Scientists at Battelle Pacific Northwest Laboratory are investigating a process for the separation of Am and Cm from the bulk of fission products (especially lanthanides) by solvent extraction that does not involve additives other than HNO_3.[5]

Since the extent to which various pathways to man's environment reduces the risk due to long-lived nuclides is not completely established, permissible concentrations of long-lived isotopes in the short-lived fraction cannot be defined. Concentrations varying from one nanocurie to ten microcuries per gram have been studied. At 10^3 nCi/g, only americium will be of concern, and the separation factor required is only 6. At 1 nCi/gm, which represents the same risk as the naturally occurring radioactivity in man's surroundings, the plutonium separation requirement is about 99.9 percent. Such removal factors are greater than those attainable.

4.1.2.2 Fission Products

The extractants generally in use for the separation of fission products fall into groups of organic phosphorous,compounds, amines, substituted phenols, ketones, etc. The best known extraction process is the use of di-2 ethylhexylphosphoric acid (HDCHP) and tributylphosphate for the extraction on Sr and the rare earths at ORNL.

The amine group includes primary, secondary, and tertiary amines and quaternary ammonium salts. The only fission products extractable by primary amines are Ru, Zr, Tc, and the rare earths. Tertiary amines used for the isolation of Ru include trialkylamines with chain lengths of six to nine carbon atoms. Dipicrylamine is used for the separation of cesium.

The use of ketones has been sporadic, such as the use of a mixture of thenoyltrifluoracetone (TTA) and tributylphosphate in CCl_4 for the extraction of Sr. Other extractants such as carboxylic acids are also in use. For example, naphthenic acid (which is ten times cheaper than HDCHP) is used in the Soviet Union in connection with the isolation of Sr and Y from neutral or alkaline solutions and extraction of Zr, Nb, Ru, Cs, and Pm.

4.1.3 Ion Exchange

The ion exchange method has several advantages such as simplicity of operation and equipment, and the possibility of using multistage arrangements. There are also drawbacks such as the slowness of the process, large volume used for elution, and unsuitability for use with uncharged substances or with colloids [e.g., polyantimonic acids $[H_3Sb_3O_5(OH)_8]_3$ or $(H_5Sb_5O_6(OH)_{18})$].

4.1.3.1 Actinides

Ion exchange methods for actinide separation are still at the laboratory stage. It has been shown that Am and Cm can be partitioned by the use of two ion exchange steps[4], with recovery capability 99.9 percent or greater. First the lanthanides, actinides, and some of the other fission products are sorbed on a cation exchange resin column and selectively eluted with HNO_3. The actinides and lanthanides are then separated by cation exchange chromatography on a second column. There are some problems yet to be resolved such as the conversion of actinide-bearing ion exchange resins to forms suitable for waste disposal, and the treatment of the waste streams generated in the chromatographic separation.

4.1.3.2 Fission Products

There are many hundreds of cation and anion exchangers being produced with different selectivities for particular ions. Some ion exchange resins of the organic synthetic type include hydroxyisobutyric acid, lactic acid, ethylene diamine tetracetic acid (EDTA), hydrazinediacetic acid (HDA), and hydroxyethyl ethylene diamine triacetic acid (HEDTA).

Synthetic resins undergo radiation damage accompanied by gradual reduction in capacity. Inorganic substances (such as hydroxides, salts of acids with multivalent metals, insoluble ferrocyamids, alumino silicates, etc.) do not have this drawback. Following are a few well known processes. MnO_2 has been used for the purification of Pm. Polyantimonic acid ($[H_3Sb_3O_5(OH_8]_3$ has been used as a selective sorbent for Sr. Zirconium phosphate, which is a well studied product, is used for the sorption of Cs. Salts of heteropolyacids, such as $(NH_4)_2HPMo_{12}O_{40}$, used in packed columns easily take up heavy alkali elements. Alumino silicates, which can be divided into clays and zeolites, are highly resistant to radiation damage. Clays, which are cheap and abundant, are used mainly in connection with the treatment of low and medium activity wastes. A large number of zeolites have been used for isolating Cs, Sr, Y, Ce, Ru, and other medium A elements.

4.1.4 Precipitation Methods

Precipitation methods make use of the low solubility of certain compounds. They date back to the days of Mme. Curie and Hahn and Meitner and are therefore well established. However, when applied to high level waste they entail the problem of remote handling of solids. They may best be used in conjunction with solvent extraction and ion exchange.

Methods for obtaining crude concentrations of Pu, Am, and Cm by oxalate precipitation have been developed at EURATOM in Ispra, Italy. Multiple stages of this type of precipitation have resulted in essentially complete removal of the Am-Cm mixture.

The separation of actinides from high-level waste solutions as hydrous oxides or associated hydroxyphosphates through the hydrolysis of urea or hexamethylenetetramine is being attempted in several labs in the U. S. and Germany.[6] This method, known as homogeneous precipitation, has the advantage that the reagents would not contribute to the volume of the high-level wastes. It also avoids the effects of introduction of a variety of other chemical substances.

The insolubility of sulphates of alkaline earths, oxalates of rare earths, and of double salts (such as alums) of alkali metals makes precipitation a very useful procedure for such fission products. For example, the best known method for isolation of Cs is the precipitation of $CsAl(SO_4)_2 \cdot 12H_2O$ (cesium aluminum sulphate). The heteropoly acids with heavy alkali elements form slightly soluble salts, e.g., phosphotungstic acid $H_3PW_{12}O_{40}$ or phosphomolybdic acid $H_3PMo_{12}O_{40}$. Ferrocyanides are another type of material used to take up alkali metals, especially Cs.

Coprecipitation is used in the isolation of Sr and rare earths (e.g., Ce, Pm). The former is precipitated with $PbSO_4$. The rare earths are precipitated as a double sulfate with sodium.

4.1.5 Individual Nuclides

The separation of ^{85}Kr, ^{90}Sr, ^{93}Zr, ^{99}Tc, ^{129}I, and ^{137}Cs are of particular interest due to their health effects, long half-lives, and/or difficulties of containment for long periods.

Krypton-85. A spent fuel reprocessing plant with a daily capacity of five tons of fuel produces 35,000 curies of ^{85}Kr per day. It would be desirable to keep dilution of the gas to a mimimum, therefore, the free space for cutting and dissolving of fuel parts is kept very small. Because ^{85}Kr is a noble gas, there is no necessity for chemical separation; physical separation methods include adsorption on solid materials and in liquids, low temperature distillation and diffusion. A more simple approach would be adsorption on activated charcoal or molecular sieves at laboratory temperature.

Strontium-90. The main emphasis for separating and refining Sr has been on precipitation or coprecipitation methods using a carbonate, an oxalate or lead sulphate; ion exchange methods based on the use of organic resins and inorganic synthetic materials; and extraction methods involving the use of the di-2 ethylhexyl phosphoric acid (HDEHP).

Zirconium-93. The oldest method of separating zirconium is based on sorption with silica gel and elution with oxalic acid, which forms a soluble complex with zirconium. Extraction with HDEHP or TBP is another possibility. Ion exchange on resins with complex functional groups has also been found feasible.

Zirconium, along with niobium, is obtained as a precipitate in alkaline wastes from the Purex process. In acid wastes these elements are partly found in the solution and partly adsorbed to solid siliceous deposits from which considerable amounts can be extracted by leaching.

Technetium-99. The principal starting material for obtaining technetium is alkaline Purex wastes from which Cs has been isolated. In acidic and alkaline solutions, especially in the presence of oxidizing agents, Tc is isolated as a pertechnate TcO_4. Technetium can also be a by-product in the preparation of UF_6 from reprocessed uranium. It is separated from UF_6 by adsorption in MgF_2, and is then refined by anion exchange or solvent extraction by a tertiary amine.

Iodine-129. Iodine-129 is a volatile radionuclide released during spent fuel reprocessing. The use of silver- and lead-exchanged zeolites for recovery from the reprocessing off-gases and storage of I-129 is now being studied at Idaho National Engineering Laboratory. Both collection and fixation of iodine are accomplished in the same process. About 1.5 cubic meters of lead-exchanged zeolite will be required annually to collect the iodine generated by a plant which reprocesses five tons of fuel per day. Immobilization of iodine in cement and glass is also being attempted.

Cesium-137. As a well-known gamma- and beta-ray energy standard, the separation and purification of ^{137}Cs has been done for a very long time. Co-crystallation of Cs with $Al_2(SO_4)_3$ was developed at Oak Ridge in the 1940s. The solution containing Cs is saturated at 80°C. with ammonium alum and cooled to 15°C. Crystals of this material are separated. At Hanford, Cs is obtained from alkaline wastes which are passed through a bed filled with aluminosilicate. Maximum selectivity for the uptake of Cs from a solution containing $NaNO_3$ or $NaNO_2$ is achieved at low temperatures.

The use of heteropoly acids is well suited for obtaining Cs from highly acidic waste solutions. The process has been tested in the U.S., U. K., and France. Cs is selectively absorbed by salts of multibasic acids of readily hydrolysable elements, including Zr, Tc, Sr, U, Th, and Ce salts of phosphoric, molybdic tungstic, antimonic and arsenic acids. Hexafluorophosphate, tetrafluorophosphate, and hexafluoroarsenate are among the extracts of Cs which have been tested.

4.1.6 Other Methods of Partitioning

A technique that is being pursued with some success at the Lawrence Livermore Laboratory is the chemical separation of transplutonium elements from the chemically analogous lanthanides.[4] This technique uses the formation of stronger complexes by virtue of the farther spatial extent of the 5f electron orbits of actinides in comparison to the 4f electron orbits of the lanthanides.

Partitioning of actinide elements from high-level wastes using laser photochemical separation is being evaluated at the Brookhaven

National Laboratory.[5] This process involves reactions that a molecule undergoes subsequent to electronic excitation by a light quantum. A general survey of the photochemical spectral region is required to determine the feasibility of introducing light into the complex process mixtures and to determine whether there are appropriate numbers of wavelengths to carry out selective photochemical reactions. If successful, this technique can be used for the partitioning as well as fractionation of the individual actinides and fission products.

4.2 Environmental and Health Considerations

The full range of the environmental impacts of applying partitioning and fractionation techniques to radioactive waste is not easy to assess because the techniques are not yet well established. It is expected that the design and construction of nuclear fuel cycle facilities using partitioning and fractionation of waste would, at the earliest, be in the 1990s. The time of implementation is dependent upon several things: a decision to proceed with spent fuel reprocessing, without which partitioning and fractionation cannot occur; the establishment of a need, for example, the commitment to a disposal concept requiring this partitioning of waste; and the rate to which research and development is funded.

The implementation of a partitioning and fractionation technology will be dependent upon the balance of the positive and negative impact on the environment and the health effects. Advancement of this technology is necessary for certain alternative radioactive waste disposal methods, in particular transmutation and extraterrestrial disposal. These alternative disposal methods are positive contributions to the extent that, singly or in combination, they reduce the risk to the environment and society both for the present and future generations.

Partitioning and fractionation will increase the steps in the handling of radioactive waste and thus will increase the radiological risk. As an adjunct to partitioning and fractionation of waste:

- The total volume of waste to be handled increases due to the chemical process involved;
- The quantities of low-level radioactive waste and contaminated facilities and equipment to be treated and disposed of increase;
- There are usually some small releases of radioactive materials and pollutants to the environment;
- There is an increased risk in occupational exposure of workers;
- Additional transportation with associated risks may be required;
- The potential for accidents will be increased.

Quantification of the potential environmental and health effects is not possible with the information available and estimation of these effects is not within the scope of this study. However, it is reasonable to assume that the impacts would be less than those from spent fuel reprocessing.[8,9,10] In the context that reprocessing will be acceptable after consideration of environmental, health, and political factors, it can be anticipated that partitioning and fractionation will also be acceptable. In the final assessments of the alternative disposal methods, those methods requiring partitioning and fractionation must include the associated impacts in the benefit and effects evaluation.

4.3 Economic Impact

The cost of partitioning high-level waste into a long-lived and a short-lived fraction will certainly increase the cost of nuclear fuel processing. The estimate made so far has been preliminary because many of the techniques are still at the laboratory level. The cost depends on the degree of separation desired and the number of elements which must be separated from the short-lived fraction.

The most conservative estimate is that in BNWL-1907,[2] where the cost of separation to an actinide concentration level of 1000 nCi/gm is set at $4/ton of uranium. The corresponding figures for 100, 10, and 1 nCi/g are $1,400, $3,900 and $4,200, respectively. These figures "probably are significantly low" according to the authors.

Another cost estimate was made on separating 99 percent of the actinide elements only from high-level waste. The process was developed by Koch, et al, in Germany. In this process, the volume of concentrated high-level waste per unit mass of irradiated fuel is about seven times less than that of the feed to the reprocessing plant. Accordingly, the basic reprocessing cost ($35,000/ton) was reduced by a factor $(x)^{1/2}$ because of the reduction in plant size. The resulting cost was further modified by comparing the number of process cycles for partitioning to the number required for reprocessing. The total cost estimates are as follows:[15]

	Cost/tons
Actinides plus 1% of fission products	$ 10,000
Actinides less U + 1% of fission products	$ 15,000
Actinides less U + 0.1% of fission products	$ 20,000

A still higher estimate for actinide partitioning has been quoted by Brown and Goldstein.[5] It is claimed that actinide partitioning cost will be comparable to nuclear fuel reprocessing costs which will be $324/kg U. Evidently, there is a wide discrepancy among the three estimates of at least two orders of magnitude. The conclusion is that at the present time, accurate predictions of the cost of partitioning are not possible.

What can definitely be said is that actinide partitioning will increase the cost of electric power and the cost of waste management research. Additionally, there are other comparable long-term hazards such as the low-level solid wastes generated at the fuel fabrication facilities, where 0.5 percent of the processed Pu and U are lost. It is indeed hard to compute the economic aspects of such problems.

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5.0 TRANSMUTATION

One of the alternatives being considered for the management of long-lived radioactive wastes is to transmute them into stable or short-lived radioactive or fissionable isotopes. If this is feasible, the quantity of waste containing long-lived radionuclides could be reduced significantly, and the time required for isolation of the waste shortened.

5.1 Transmutation Concepts

The process of transmutation is accomplished by any of the following devices:

- Particle accelerators;
- Thermonuclear or fission explosives;
- Fusion reactors;
- Fission reactors.

Each type of device has to be judged on the basis of certain criteria including overall energy and waste balance and the rate of transmutation. A favorable overall energy balance means that the energy required to dispose of the waste should be less than the energy furnished by the nuclear reactor which produced the waste, preferably by an order of magnitude or better. A conceivable exception would be when the era of nuclear fission power comes to an end and there are other plentiful energy sources available which can be economically used for the disposal of the fission power wastes. The criterion of overall waste balance is self evident: the waste disposal program should not create more hazardous waste than it removes. This is not as trivial as it first appears. The process of transmutation in some cases is similar to the original process which created the waste. A successful transmutation rate would be greater than the natural decay rate of the nuclide. More precisely, the product of the particle flux (ϕ) which induces the transmutation and the cross-section (σ) for the transmutation process should be much greater than the natural decay constant of the nuclide (λ), i.e., $\phi\sigma >> \lambda$.

5.1.1 Particle Accelerators

At least four accelerator transmutation methods are conceivable: (1) direct bombardment by charged particles of several hundred MeV energy; (2) coulomb excitation in order to augment the β-decay rates; (3) photon transmutation using electron bremsstrahlung; and (4) use of neutrons released as a result of spallation by high energy particles.

5.1.1.1 *Direct Bombardment by Charged Particles*. The direct nuclear reaction of charged particles from accelerators is not particularly attractive for radioactive waste transmutation. Most of the long-lived fission products are intermediate or high atomic number nuclei. Proton penetration for such nuclei requires energy of several tens or hundreds of MeV. It has been estimated that nuclear reaction with direct bombardment by charged particles expends at least five times the energy in transmuting the waste than was acquired in creating it.[1]

5.1.1.2 Coulomb Excitation. Beta-decay from certain metastable nuclear excited states proceeds more rapidly than that from ground states. This situation applies only in certain exceptional cases. For example, the 10.8 year Kr-85 has a metastable state at 310 keV which decays with a half-life of 4.4 hours. Unfortunately the cross-section for Coulomb excitation is so small that the energy requirement is higher by three orders of magnitude than the nuclear fission energy which produced the waste.[1]

5.1.1.3 Photon Transmutation. Electrons accelerated to several tens of MeV produce a shower of photons from bremsstrahlung, but the yield of photons is too small and the energy required is found to be at least two orders of magnitude greater for the actual transmutation of waste nuclei than the energy produced during the creation of the waste.[1]

5.1.1.4 Spallation Neutrons. High energy acceleration with proton energy greater than 1000 MeV could provide a continuous source of neutrons by spallation in suitable targets (e.g., Pb-Bi). After moderation in a suitable medium, thermal neutron fluxes up to 10^{16} n/cm^2 sec can be expected and can be used for transmutation. The energy required to transmute one fission product nucleus such as Cs-137, Tc-99, or Sr-90 was estimated to be between 23 and 110 MeV.[2] Thus this method would at best be marginal in satisfying the energy balance criterion. With a proton beam power of 65 MW, it is estimated that two spallation accelerators are needed to handle the inventory of the above mentioned isotopes. However, at a flux of 10^{16} neutrons/cm^2 sec, it takes 14 years to eliminate 99 percent of ^{90}Sr and 80 years for ^{137}Cs.[3] The radioactive contamination caused by proton interaction with structural materials, the lead target, etc., may create more wastes than it can transmute, but these are expected to be short-lived.

Another possibility with high energy accelerators is to use the radioactive fission product as the target for the protons. A study team of the Japanese Industrial Forum has speculated that 85 ^{137}Cs nuclei could be transmuted per incident proton.[1]

5.1.2 Nuclear Explosives

Transmutation using fission and thermonuclear explosive devices has been evaluated as technically feasible.[4] The procedure is to partition the actinides and to lower them into a drilled hole along with the explosive device, seal the hole, and set off the device. The neutrons produced in the explosion transmute the waste. It is estimated that an average of 3.5 one-hundred kiloton thermonuclear detonations would be required annually to transmute the Np, Am, and Cm produced every year in a 1000 MWe light-water reactor. It should be borne in mind that the fission products resulting from the actinide transmutation and the unconsumed fissile material of the device will remain in place along with those resulting from the nuclear explosion. Transmutation of long-lived fission products from each 1000 MWe LWR requires more than 11 one-hundred kiloton detonations. The concept of transmutation by means of nuclear explosive devices is not considered practical because of the inordinate number of explosions needed to cover the nuclear power capacity of the world.

5.1.3 Fusion Reactors

Fusion reactors potentially have very high neutron flux levels (10^{15} - 10^{16} neutrons/cm^2 sec). The high energy neutrons produced in fusion reactors can be used directly to cause neutron induced reactions or thermalized for capture in fission processes.

A study of actinide transmutation in the blanket of a conceptual thermonuclear fusion reactor has been made by Wolkenhauer, Leonard and Gore for both deuterium-deuterium (D-D) and Deuterium-Tritium (D-T) reactions. The flux of the neutrons from the plasma reactions could be augmented by a factor of 2.5 by having beryllium in the blanket (using the reacton $^9Be + n \rightarrow 2\ ^4He + 2n$), thus, fluxes up to $3x10^{16}$ neutrons cm^{-2} sec^{-1} could be realized, which is about 1000 times greater than in an LWR. Fluxes of this order of magnitude raise the possibility of transmuting not only actinides but also fission product nuclides such as Kr-85, Zr-93, Tc-99, and I-129 for which there is no practical way of transmutation using fission reactors (see below). In addition to capture and fission processes, there are other possibilities such as (n, 2n), (n, 3n) and (n, charged particle) reactions for nuclides such as Np-237, Pu-237, and Am-234. Since a sustained controlled thermonuclear reaction has not yet been achieved, use of this technique for waste management has to await a breakthrough in controlled thermonuclear reactor technology.

If, and when, fusion reactors become commercial, it is very likely that fission reactors will no longer be built, thus the fusion reactors will only have to transmute whatever inventory of fission products and actinides are left. In the long run, transmutation by fusion reactors may become unnecessary.

5.1.4 Fission Reactors

The suggestion to use neutrons from fission reactor neutrons to to transmute radioactive waste was made as early as 1964.[6] A study by Claiborne[7] made at the Oak Ridge National Laboratory is perhaps the most extensive study of the subject to date. It is the general consensus of this and later studies that transmutation of actinides in fission reactors is technically feasible. Kubo,[8] and Kubo and Rose,[9] extended Claiborne's work and have shown that actinide recycling in thermal reactors is not only technically feasible, but is an attractive waste management concept. In the scheme visualized by Claiborne, the chemical processing of the irradiated fuel rods is separated into three parts: (1) 99.5 to 99.9 percent of uranium and plutonium stored or recycled because of their fuel value; (2) fission products and approximately 0.1 to 0.5 percent heavy elements; and (3) 99.5 to 99.9 percent actinides other than U and Pu. Uranium and Pu are then recycled into the fresh fuel by adding uniformly to every rod of a 3.3 percent enriched UO_2 fuel for a PWR.

There are a number of parameters which have a bearing on the feasibility and effects of actinide transmutation in fission reactors:

- mass and composition of actinides being recycled;
- the rate at which the recycled actinides are fissioned in the various types of fission reactors;
- the effect of the recycled actinides on fission reactor criticality and reactivity;
- the effect of the recycled actinides on fuel fabrication, shipping reprocessing, etc.

It has been estimated that a pressurized water reactor producing 1000 MWyr(e) electric produces about 22 kg of actinide waste after recovery of Pu and U, of which Np, Am and Cm constitute 70, 23 and 6 weight percent, respectively. These are typical values and the exact composition depends on the reactor characteristics and the recovery techniques.[10]

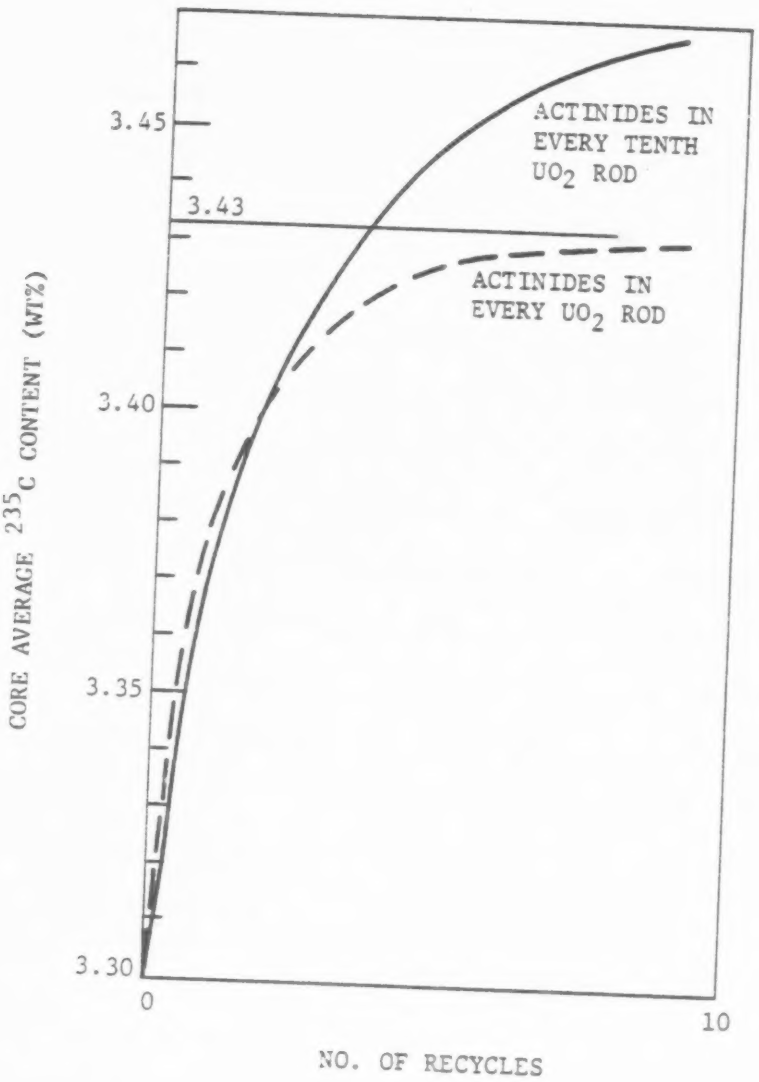
5.1.4.1 Light-Water Reactors. A pressurized water reactor using U-235 and U-238 fuel with 3.3 percent U-235 enrichment has been considered as a typical transmuting reactor in the study by Claiborne.[7]

Sustained recycle of the actinides in a pressurized water reactor results in an equilibrium mass of about twice that produced per year without recycling.[11] It has been estimated that a typical actinide transmutation rate is about 6 percent for each year that the actinides are in the transmutation reactor.

The introduction of actinides in the fuel rods affects the neutronic behavior of the reactor. The infinite multiplication factor k_∞, which is the ratio of the neutron production rate to the neutron destruction rate assuming no leakage (reactor of infinite size), is one criterion for neutronic behavior of reactors. The effect on k_∞ is a function of the time spent by the actinide in the reactor, but on the average the effect is to decrease k_∞ by 1 percent.[9] This, of course, seriously affects the reaction design, the economics and the uranium resource utilization.

The incorporation of the actinides into the fuel elements can be done uniformly, or in certain selected fuel rods. The mixture of the actinide isotopes into the fuel elements will present some fabrication problems due to decay heat, gamma-ray dose rate, and neutron emission rate, thus causing additional fabrication costs.

The increase in uranium enrichment required to achieve the same energy output as for 3.3 wt percent UO_2 fuel without actinides is shown in Figure 5-1 for two strategies. In the first strategy, the actinides are distributed uniformly in all the fuel rods, and in the second they are concentrated into every tenth fuel rod. Assuming one-third of the core is discharged every year (corresponding to an average burnup rate of 22,000 MWd/ton), the enrichment of the fuel must be increased from 3.3 to about 3.43 percent for the first strategy. Recycling actinides in ten percent of the rods increases the demand on uranium enrichment to 3.47 percent on the average. The reason for the higher enrichment requirement is that higher



Source: Kubo and Rose, Reference 9.

FIGURE 5-1
ENRICHMENT REQUIREMENTS FOR ACTINIDES RECYCLE

concentration causes additional self-shielding of the resonances, requiring a larger inventory of each isotope before the burnup rate equals the production rate. Table 5-1 compares the actinide inventories for the two recycle strategies. It shows that actinide inventory is not reduced as much by the recycle of actinides in a few rods as in the recycle in every fuel rod.

5.1.4.2 Fast Neutron Reactors. It was recognized early in the recycling studies that fast neutron reactors would cause a faster burnup of the actinides than thermal reactors. There are two very obvious reasons for this: 1) the fission-to-capture ratio* is generally higher for fast reactor neutrons; and 2) the flux of the fast reactors is typically 5×10^{15} n/cm^2 sec as against 3×10^{13} for LWRs. The combination of flux and cross-section rates results in higher fission rates of the actinides compared to thermal reaction (see Table 5-II).

Beaman and Aitken[12] tried to determine the equilibrium cycle condition for a recycle scheme involving one 1200 MWe Liquid Metal Fast Breeder Reactor (LMFBR) and three LWRs of comparable power, using only the LMFBR as the transmuting reactor. In their calculations, they assumed a two-year period for reprocessing and fabrication between the time of discharge from the reactor and time of loading in the LMFBR. The batch stays in the LMFBR for 402 days. Table 5-III

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*Fission to capture ratio is the ratio of the number of neutrons resulting in actinide fission to the number of neutrons absorbed by the actinide nucleus resulting in isotopic transmutation to another isotope of the actinide.

TABLE 5-I

COMPARISON OF ACTINIDE INVENTORIES FOR TWO RECYCLE STRATEGIES

Part A - Actinides Recycled in All Rods

	Actinide Inventory (Gms/MT of Heavy Metal)								
Recycle No.	Np237	Am241	Am242	Am243	Cm242	Cm243	Cm244	Cm245
0	521.28	64.61	.58	78.63	7.89	.12	22.90	1.07
1	703.92	76.41	.77	100.41	9.95	.28	76.66	5.75
2	921.49	77.51	.79	104.67	10.15	.33	116.20	9.63
3	993.84	77.81	.80	105.12	10.15	.34	139.81	12.02
4	1031.99	77.95	.80	104.94	10.15	.34	152.96	13.37
5	1052.15	78.03	.80	104.72	10.15	.34	160.08	14.10
6	1062.81	78.06	.80	104.58	10.14	.34	163.88	14.49
7	1068.45	78.00	.80	104.49	10.14	.34	165.90	14.70
8	1071.44	78.10	.80	104.44	10.14	.34	166.97	14.81
9	1073.02	78.10	.80	104.42	10.14	.34	167.53	14.87

Part B - Actinides Recycled in One Rod in Ten

	Actinide Inventory (Gms/MT of Heavy Metal)							
Recycle No.	Np237	Am241	Am242	Am243	Cm242	Cm243	Cm244	Cm245
0	521.28	64.61	.58	78.63	7.89	.12	22.90	1.07
1	811.78	73.37	.71	103.43	9.68	.27	76.55	5.96
2	994.46	74.98	.75	111.98	9.90	.32	120.63	10.44
3	1118.27	75.85	.78	115.43	9.98	.34	152.73	13.75
4	1192.93	76.31	.79	116.81	10.01	.35	175.55	16.05
5	1253.06	76.68	.80	117.66	10.04	.35	193.78	17.88
6	1299.96	76.96	.81	118.01	10.06	.36	203.55	18.95
7	1334.71	77.15	.81	110.16	10.07	.36	208.71	19.57
8	1360.02	77.27	.82	118.22	10.08	.36	211.28	19.91
9	1379.70	77.35	.82	118.26	10.09	.36	212.44	20.10

Source: Kuba and Rose, Reference 9

TABLE 5-II

ACTINIDE REACTION RATES IN FAST AND THERMAL REACTORS
(Reactions/sec/Atom)

		Fast Spectrum*		Thermal Spectrum**	
Isotope	Half-Life, Years	Fission Reaction Rate	Capture Reaction Rate	Fission Reaction Rate	Capture Reaction Rate
Np^{237}	2.14×10^6	2.2×10^{-9}	1.03×10^{-8}	6.18×10^{-12}	4.9×10^{-8}
Am^{241}	433	2.7×10^{-9}	2.35×10^{-8}	6.18×10^{-10}	1.38×10^{-7}
Am^{242m}	152	4.7×10^{-8}	9.69×10^{-9}	1.49×10^{-7}	1.33×10^{-7}
Am^{243}	7370	1.39×10^{-9}	4.5×10^{-9}	1.55×10^{-11}	3.18×10^{-8}
Cm^{244}	17.9	3.47×10^{-9}	2.77×10^{-9}	4.02×10^{-10}	5.9×10^{-9}

*Average Total Flux = 6.93×10^{15} in Core Zone 1
**Average Total Flux = 3.09×10^{14}

Source: Raman, Reference 11

TABLE 5-III

ACTINIDE RECYCLE FROM ONE 1200 MWe LMFBR AND THREE 1200 MWe LWR's

Cycle No.	Total Actinides[a]	Total Pu[b]	Total Actinides + Pu	Total Actinides if not recycled
2	1.47 + 2	3.05 + 1	1.77 + 2	2.26 + 2
4	2.28 + 2	4.42 + 1	2.72 + 2	4.52 + 2
6	2.75 + 2	5.05 + 1	3.25 + 2	6.78 + 2
8	3.02 + 2	5.36 + 1	3.56 + 2	9.04 + 2
10	3.19 + 2	5.51 + 1	3.74 + 2	1.13 + 3
12	3.30 + 2	5.58 + 1	3.86 + 2	1.36 + 3
14	3.37 + 2	5.62 + 1	3.93 + 2	1.58 + 3
16	3.41 + 2	5.63 + 1	3.92 + 2	1.81 + 3
18	3.44 + 2	5.64 + 1	4.01 + 2	2.03 + 3
20	3.47 + 2	5.66 + 1	4.03 + 2	2.26 + 3
22	3.48 + 2	5.66 + 1	4.04 + 2	2.49 + 3
24	3.49 + 2	5.66 + 1	4.05 + 2	2.71 + 3
26	3.50 + 2	5.66 + 1	4.06 + 2	2.94 + 3
28	3.50 + 2	5.66 + 1	4.06 + 2	3.17 + 3
30	3.51 + 2	5.66 + 1	4.08 + 2	3.39 + 3

a) Actinides include: Np, Am, Cm, Bk, Cf

b) Pu results from Np neutron capture or decay of higher atomic number isotopes

Source: Beaman and Aitken, Reference 12.

lists the total weight of actinides remaining after a specified number of cycles, and the total weight which would be accumulated if the actinides are not recycled. The number of cycles required for equilibrium of a particular isotope increases with increasing atomic weight because of the production of higher atomic number isotopes by neutron capture in the lower atomic number isotopes.

During their lifetime (assumed to 40 years), the four reactors would have produced about 3620 kg of actinides; with recycling this would be reduced to 690 kg, thus reducing the actinide quantities by a factor of 5.2 over the life-time of the reactors. If the reactors are replaced by another generation of comparably powered reactors and the recycling is continued, the equilibrium concentrations will remain the same and a reduction factor of more than 10 is achieved over a period of 80 years.

Actinide recycle might affect the transmuting reactor in several ways. These could include the increase in fissile inventory, reactivity of the core, and breeding ratio. These effects were explored by Beaman and Aitken[12] by a comparison between the "reactivity worths" of standard fuel assemblies and target recycle assemblies defined as:

$$\sum_{i=1}^{n} N_i \ \nu_i\sigma_{if}-\sigma_{ia}$$

Where N_i is the atom density of the i^{th} nuclide, ν_i, the average number of neutrons it emits per fission, σ_{if}, σ_{ia} are the one-group microscopic fission and absorption cross-sections in the number of reactive materials. The actinides, because of their larger absorption cross-section in comparison with U-238 (which forms the bulk of the fuel assembly), have a negative worth, but $\nu\sigma_f$ is relatively large for the actinides and this almost compensates for the absorption. The decrease in reactivity worth is only slight for a 50-50 U-238 actinide mix replacing an equal number of standard fuel assemblies. The worth of the core can be restored in the "worst" case situation by addition of plutonium, amounting to about 3.4 percent. Such an increase in the plutonium and the decrease in U-238 which has been replaced by actinides causes a decrease in the breeding ratio of the reactor. It has been estimated that an equilibrium cycle load of actinides will decrease the breeding ratio by a rather modest amount of 1 percent.

Actinide recycling can also cause power peaking problems in a fast reactor. A fuel assembly completely loaded with recycle actinides can produce about twice as much power as a standard fuel assembly and could cause severe heat transfer and reactivity problems in the reactor. One way to avoid this problem is to mix the actinide with a dilutent. The most obvious dilutent is U-238, not only because it is plentiful but because it contributes to the breeding and minimizes the heat transfer effects. A logical choice is an assembly of 50 percent U-238 and 50 percent actinides. It has been estimated that the power output of such a fuel assembly after the attainment of actinide equilibrium varies from 8.2 MWth to 9.5 MWth, whereas the standard fuel assembly produces between 8.2 and 8.5 MWth, and this is judged to be a reasonable match.

Table 5-IV, by Beaman and Aitken,[12] lists five possible actinide recycle schemes. Each subsequent scheme has a greater safety margin and involves higher costs than the previous one. The only exception is Scheme 5, which relaxes the requirement on lanthanide fission product separation.

5.1.4.3 Thorium-Uranium Reactors

Light-water reactors using a mixture of U-235 and U-238 are the major types of thermal reactors that are in commercial use in the United States. Neutron capture by U-238 results in production of Np, Pu, and higher elements which contribute to the bulk of the actinide problem. A possible alternative would be reactors which use Th-232 as the fertile material and U-233 as the nuclear fuel. In such a reactor, the production of nuclides with mass numbers above 237 is negligible because of the large number of neutron caputures necessary to produce them. The recycling of actinides in such a reactor has been the basis of a study by Raman, Nestor, and Dabbs.[13] The Np, Am, Cm, and higher isotopes, together with 0.5 percent of the U and Pu isotopes from a U-235 and U-238 reactor, were considered as wastes to be recycled in a 1000 MWe pressurized water reactor which uses the U-233 and Th-232 cycle.

In 60 years, which corresponded to ten recycling periods, negative buildup gradients were established for all isotopes except the 5550 year Cm-246 and the 2.55 year Cf-252. Both of these are spontaneously fissionable materials and therefore require additional care in transportation and fuel processing.

ACTINIDE RECYCLE SCHEMES

Scheme	Initial Reprocessing	Fabrication	Actinide Irradiation	Reprocessing of Pins Containing Recycled Actinides
1	Remove U, Pu, Np, Am, Cm, Bk, and Cf from spent fuel	Fabricate pins ccontaining U, Pu, Np, Am, Cm, Bk and Cf	In all fuel pins of a LMFBR	Similar to initial reprocessing
2	Reprocess as in 1 above and further remove curium for storage	Fabricate as in 1 above without curim	Curium allowed to decay; irradiate after radiation levels have fallen	Similar to initial reprocessing
3	Reprocess spent fuel such that the U and Pu are separate from the Np, Am, Cm, 3k, and Cf	Fabricate fuel pins containing U, and Pu; fabricate target pins containing Np, Am, Cm, Bk, and Cf, and a possible diluent	In target pins initially containing only Np, Am, Cm, Bk, Cf, and a possible diluent	i) Reprocess target pins separately from fuel pins ii) Mix material from target pins with material from spent fuel; reprocess in a manner similar to initial reprocessing
4	Reprocessing spent fuel such that U, Pu, and Np are separate from the Am, Cm, Bk, and Cf	Fabricate fuel pins containing U, Pu, and Np; fabricate target pins containing Am, Cm, Bk, Cf, and a possible diluent	Np irradiated in fuel pins; Am; Cm, Bk, and Cf irradiated in target pins	i) Reprocess target pins separately from fuel pins ii) Mix material from target pins with material from spent fuel; reprocess in a manner similar to initial reprocessing
5	Reprocess as in 3 or 4 above, carrying some of the lanthanide fission products with the Am, Cm, Bk, and Cf	Fabricate as in 3 or 4	Irradiate as in 3 or 4	Reprocess recycle pins as in 3 or 4

5.1.4.4 Actinide Cross-Sections

The quantitative prediction of various nuclei produced, transmuted, and fissioned in reactors is necessary for systematic management of actinide wastes. Such predictions are made with the aid of special computer programs which use as input the relevant cross-sections for capture, fission, or other processes that are caused by the neutrons. Lacking detailed experimental values at the present time, most calculations utilize "effective values" in the thermal, resonance fast neutron regions.

Several laboratories in the United States have cross-section measurement programs for various actinide nuclei. The Oak Ridge National Laboratory High Flux Isotope Reactor has been used to obtain the cross-sections for the heavier actinides in the thermal and resonance regions. The Idaho Experimental Breeder Reactor (EBR II) is being used to provide integral cross-section data by the irradiation of purified samples of the isotope for several years and subsequent mass spectrometric and radiometric analysis of the sample after a certain cooling-off period. The Los Alamos Radiochemistry Group has also made integral cross-section measurements in critical assemblies using activation and fission chamber techniques.

Cross-section measurements can also be made with the aid of accelerators. The electron linear accelerators provide a versatile pulsed source of neutrons whose energy can be measured to a fair degree of accuracy by time-of-flight techniques. The Lawrence Livermore Laboratory Linear Accelerator is being used for cross-section measurements in the neutron energy range 0.1 to 30 MeV on several isotopes of uranium, plutonium and curium.

A detailed program for the measurement of actinide cross-sections has been formulated at the Oak Ridge Linear Accelerator. Some of the proposed and current measurements have been discussed by Dabbs.[14] One of the chief difficulties with cross-section measurements is the difficulty in producing isotopically pure samples.

5.1.4.5 Fission Product Transmutation

The significant fission products that have half-lives greater than 10 years and therefore need storage for more than 100 years in order to reduce their activity by a factor of 1000 are H-3 (12.33 yrs), Kr-85 (10.73 yr), Sr-90 (29.0 yr), Zr-93 (9.5 x 10 yrs), Tc-99 (2.13 x 10^{15} yrs), I-129 (1.7 x 10^{7} yrs), and Cs-137 (30.1 yrs). Carbon-14 (5730 years) is also present from activation of impurities of nitrogen in the fuel elements. Of these, tritium and C-14 can be ruled out as candidates for transmutation because their capture cross-sections for both thermal and fast neutrons are very small, of the order of microbarns. Even at a flux of 10^{17} neutrons/cm sec, the transmutation constant is only about 10^{-13} $\sec^{-1}$ compared to the natural decay constant of 10^{-12} $\sec^{-1}$ for the relatively long-lived C-14.

Tc-99 and I-129 have the highest thermal neutron cross-sections of the remaining radionuclides of 44.5 and 34.5 barns and effective fast neutron cross-sections of 0.2 and 0.24 barns, respectively. At a flux of 3×10^{13} neutrons/cm^2 sec, reduction of the technetium activity by a factor of 1000 would require 165 years, and to 10 percent would require 55 years, corresponding to an annual reduction of 4.3 percent. Even though fast reactor fluxes are much higher, the much lower cross-section makes these time periods even longer. Thus transmutation of long-lived fission products is considered impracticable, except perhaps with high energy (>GeV) accelerators.

5.2 Environmental and Health Considerations

The topics considered so far concern the burn-out efficiency for the actinides in fission reactors, but there are other considerations. One of the main results of recycling actinides would be the augmentation of spontaneous fission activity associated with the fuel. This, along with the intense activity, is a factor in the handling of actinides for chemical separation and other processes. The neutron source strength in irradiated fuel is also important in the design of shielding and it affects the reactivity status of reactors that have been shut down (i.e., its closeness to criticality).

Recycling in thermal reactors results in the production of the spontaneously fissile nuclide Cf-252. Recycling in fast reactors produces, in addition, the fissile nuclides Cm-244 and Cf-250. The short half-life of these isotopes, namely 18 years for Cm-244 and 13 years for Cf-250, make for high specific activity.

In recycling schemes under consideration, it is often necessary to fabricate the actinide holding fuel rods without those elements whose isotopes have high neutron activity. For example, if curium is removed from the recycle scheme, neutron sources for (α, n) reactions with the fuel are reduced by a factor of 18.5 and spontaneous fission neutron sources are reduced by a factor of 3100.

Further, there is the problem of highly intense gamma ray emission from such isotopes as Am-243 and Np-239. Such large dose rates may necessitate recycling in a special small throughput remotely maintained facility.

Another potential problem regarding actinide recycling is the buildup of plutonium isotopes such as Pu-238 which is reprocessed along with plutonium fuel in discharged fuel assemblies. The high alpha activity of this nuclide may dictate the maintenance of the fabrication facility for target assemblies separate from other fuel assemblies.

Actinide transmutation necessarily requires partitioning the actinide elements from the fission products, and in many instances fractionation of individual actinide elements (or at least groups of them) from other actinides. The techniques for partitioning and fractionation were discussed previously. One significant feature of partitioning and fractionation is that they would require additional radiological protection in the fuel reprocessing plants. As the actinide elements are recycled in fission reactors, actinides of higher atomic numbers and masses are produced by the successive capture of neutrons. These higher elements decay by α, β, and γ radiation and some undergo spontaneous fission, thus increasing the radiological risks.

The need for nuclear data on the actinide elements includes those for the measurement of body burden and for the estimation of internal dose. The details should include β-decay energies, Auger electron yields, fluorescent yields (X-ray), etc. for each element produced and the daughter nuclides. All of these depend on the details of the decay scheme of each nuclide produced, which need to be well established.

The phenomenon of spontaneous fission has greater radiological consequences than the other types of decay. If spontaneous fission occurs 1 percent of the time compared to the other modes of decay, the resulting dose will be comparable to that from other modes. Over 80 percent of the dose from spontaneous fission will be imparted to the organ in which the radionuclide is deposited. In the gastrointestinal tract, however, the fission fragments do not penetrate the mucosa overlying the radio-sensitive cells, so in this part of the human body a significant portion of the dose is imparted by neutrons, β-particles and γ-rays rather than the fission fragments.

From the radiological point of view, short-lived isotopes which cause the greatest concern are the following:

	Decay mode(s)	T_{12}		Spontaneous fission cross-section (barns) if any
^{241}Pu	$\alpha,\beta,\gamma,$	15	yrs	1,110
^{243}Pu	β,γ	4.98	hrs	-----
^{242}Am	β,γ	152	yrs	3,000
^{244}Am	β,γ	10	hrs	2,300
^{244}Cm	α,γ	18.1	yrs	-----
^{250}Bk	β,γ	3.2	hrs	3,000
^{250}Cf	α,γ	13.0	yrs	-----
^{252}Cf	α,γ	2.65	yrs	3,750

Implementation of a technology for a transmutation of radioactive waste will have similar environmental and health impacts as those wastes for partitioning and fractionation of waste. In addition, irradiation targets or elements will have to be fabricated, handled, and transported. Additional facilities and waste management process steps can be expected to have some effluent releases to the environment, to increase the occupation exposure of workers, and to increase the risk of accidents. The transportation of materials from chemical separation facilities to preparation and fabrication plants and to and from irradiation facilities will require special consideration to minimize the risk to the general population.

5.3 Economic Impact

Transmutation of actinides, even though technically feasible, involves economic penalties. There are several reasons for this:

- With actinide recycling, all uranium oxide fabrication will have to be remotely handled; cost increases up to five times have been estimated for remote handling. Such cost increases can be minimized by recycling in only a small fraction of the fuel rods, say 10 percent, thus fabricating the other 90 percent without a cost penalty.

- Neutron dose rates of up to 10^{11} neutrons/sec per ton of fuel material are realized after a few recycles, primarily due to Cf-252. The transportation of these materials from the reprocessing plant to the actinide target facility involves the cost of heavy neutron shielding. This could be minimized by having the target manufacturing facility as an integral part of the reprocessing plant.

- The neutronic penalty incurred in the recycling of actinides has already been discussed. As seen before, the enrichment in the fuel rods must be raised from 3.3 wt percent to 3.47 percent for the case of recycling in 10 percent of the fuel rods, which is assumed as the reference for estimating costs.

The estimated annual incremental costs for the transmutation of actinides are listed in Table 5-V[1]. The figure of $45 million (1973 dollars) can propagate to an increase in the cost of electricity. One thousand tons of fuel corresponds to the reprocessing requirement of 33, 1000 MW PWRs per year, which at 70 percent capacity fuel will produce about 2 x 10^5 GWh electricity per year.

As shown by Beaman and Aitken,[12] the reduction in the breeding ratio is very small (1 percent) and the economic penalty is negligible.

TABLE 5-V

INCREMENTAL COST FOR TRANSMUTATION OF ACTINIDES

Component	Annual Cost/1000 Tons of Fuel* ($ x 10^6)
Partitioning	10
Fabrication	21
Enrichment	14
Total	45

*Cost in 1973 dollars.

Source: Battelle, Pacific Northwest Laboratories, Reference 1.

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6.0 EXTRATERRESTIAL DISPOSAL

Radioactive nuclear waste launched deep into space without any possibility of return to earth is permanently removed from our environment. The long-lived wastes, with half-lives of thousands to millions of years, may thus be disposed of without concern for the long-term integrity of their containers. This attractive possibility has created and sustained the interest in extraterrestrial waste disposal for the last ten years or more.

The most extensive studies were conducted by the National Aeronautics and Space Administration (NASA) with contributions from ERDA (now DOE) and were published in 1973-74.[1,2] These two studies were performed concurrently and some authors are common to both. Their work established the technical feasibility of space disposal of transuranium wastes, estimated the costs, and assessed the safety implications. The scope of the work was based on utilization of existing technologies in order to avoid any implication of unreality or a desire to promote any particular idea. This paper summarizes the results, updates the cost estimates, and further assesses risks and benefits.

The results of a more recent study of extraterrestrial disposal of radioactive waste conducted by Battelle Columbus for NASA are included to some extent in this report.[3] This latter study is only a part of several concurrent studies sponsored by NASA. When complete, this study will provide an updated assessment of the feasibility and risk of extraterrestial disposal. Since the assumptions and technical approach will be more advanced than those of references 1 and 2, they should be consulted as available.

6.1 Basis of Reference Studies

The studies referred to were based on an assumed nuclear capacity of 1000 GWe.[1,2] This is consistent with the presently projected upper limit of installed nuclear power in this time period. The estimated weight of waste accumulated after removal of uranium and plutonium by the year 2000 was estimated at 9000 metric tons (MT) of fission products and 1200 MT of actinides. The 1200 MT of actinides reduces to 300 metric tons if the separation of uranium is complete. These results compare on the low side to those of reference 4, which estimates 9,000 to 22,000 MT of fission products and 700 to 1,600 MT of transuranium products for 400 to 1,000 GWe gross installed capacity for a mixed oxide recycle (see Section 3.0). The estimates of reference 4, however, are based on the total waste produced over the 30-year plant life. All of these wastes would not be available for disposal in the year 2000. Several options for the space disposal of reactor wastes were considered in the studies:

A. Launching all wastes;

B. Removing fission products, uranium and thorium, and launching only the transuranimum elements with 1.0 percent, 0.5 percent or 0.1 percent of the fission products remaining;

C. Same as B, but with 99 percent of the curium removed.

It was evident in the early studies performed and the more recent study of reference 3 that an impractically large number of launches--thousands of flights per year by the turn of the century--would be required following option A. Similarly, by the year 2000, approximately 15,000 metric tons of spent fuel were estimated to be generated per year. Launching of this large mass is also considered impractical. Accordingly, extraterrestrial disposal of spent fuel from the "throwaway" cycle is also impractical.

Option C was considered because curium-244 with a half-life of 18 years is responsible, after removal of uraniuim and plutonium, for all but 15 to 20 percent of the actinide radioactivity and about 10 percent of the heat in 10-year old light-water reactor wastes. Removal of the curium substantially reduces the heat removal and shielding requirements thereby allowing an increase in launch payload and a corresponding decrease in cost and number of launches. It was assumed that the spent fuel would be held for at least 10 years, and possibly much longer, prior to processing. A longer period of terrestrial storage of the waste would, of course, allow the curium to decay and thus reduce the heating and shielding problems. Realizing that an optium hold time would actually be used based on costs of holding, encapsulation, and transportation (launch), the studies of curium-244 removal were not pursued in detail. The limited results of the study indicate an approximate 50 percent reduction in extraterrestrial disposal costs if the curium-244 is removed from 10-year old reactor wastes.

It should be borne in mind, therefore, that the costs presented for the cases without curium removal may be conservative.

Primary attention in the studies of extraterrestrial disposal has been given to option B and the percentages of fission products were treated parametrically in some instances. It was judged that separation technology would more nearly satisfy the 1.0 percent fission product content, so primary emphasis was given to this case although a ten-fold reduction in fission product content (0.1 percent) could provide a reduction of up to 50 percent in program costs.

Very long-lived fission products such as Zr-93, Tc-99, and the volatile radionuclides I-129 and C-14 were not considered for separation from the fission products and extraterrestrial disposal along with the actinides. The iodine fraction of the total waste is approximately 0.1 weight percent and the technetium fraction is even less. The actinide fraction consisting primarily of neptunium, plutonium, americium, and curium is approximately two percent. The chemical form and packaging that would be chosen for iodine and technetium and other long half-life fission products have not been determined. As will be evident from cost breakdowns subsequently presented, the major cost is in the transport of waste to the space destination. Extraterrestrial disposal of the long-lived fission product wastes will be costly and will have to be weighed against the advantages of reduced potential health effects to future generations.

The waste fraction for space disposal primarily discussed in the balance of this paper is the separated actinides with one percent of all fission products remaining, and uranium removed and aged ten years from reactor withdrawal.

6.2 Space Disposal Concept

The required steps in space disposal are shown in the simplified diagram of Figure 6-1. Spent fuel is withdrawn from storage, reprocessed, and partitioned into fission products and actinides with the uranium removed. Volatile radionuclides are released during reprocessing and the long half-life radionuclides I-129 and C-14 could be collected and prepared for space disposal. Uranium may or may not be separately extracted for re-use. Fission products are assumed to be prepared and disposed of by different methods. The transuranium products are processed and encapsulated, then shipped to the launching site where they are launched for space disposal.

6.2.1 Waste Capsule and Reentry Shield

The capsule and shield must provide the following:

- Integrity for the time of use up to final space disposal
- Safety in ground handling
- Shielding
- Integrity in case of accidents
- Cooling and heat transfer
- Handling and attachments
- Subcriticality

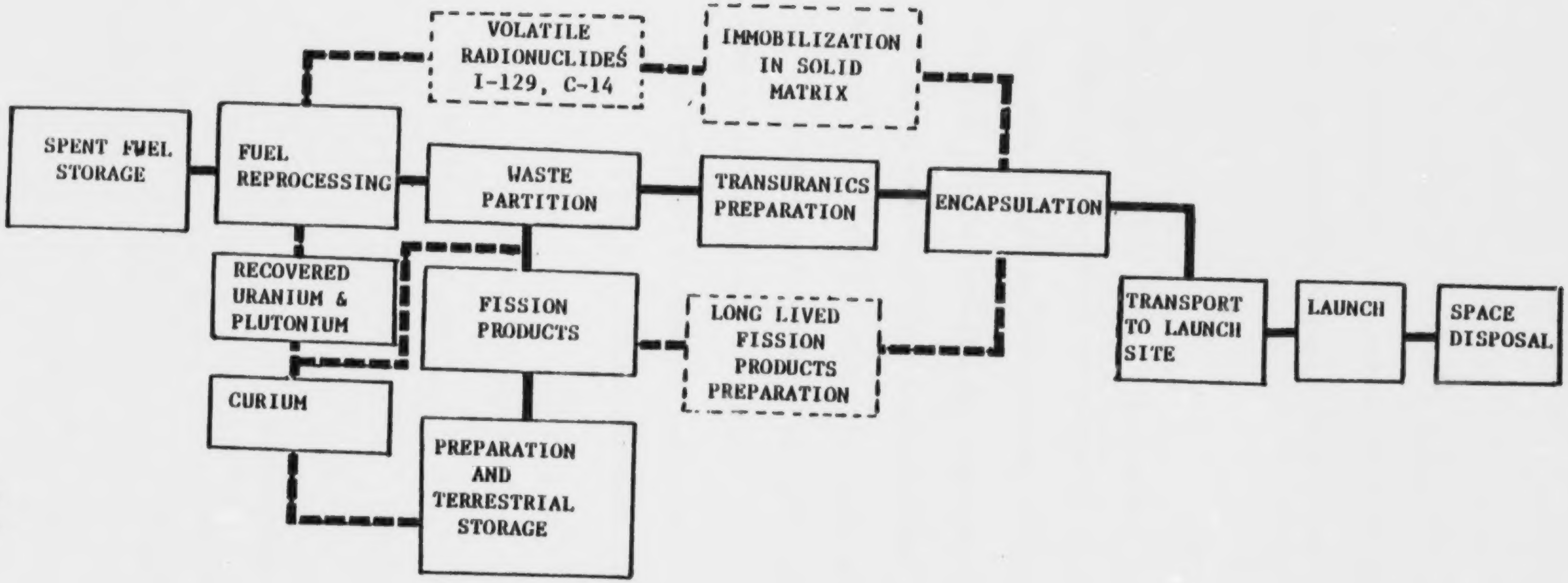
SPENT FUEL STORAGE
FUEL REPROCESSING
VOLATILE RADIONUCLIDES I-129, C-14
IMMOBILIZATION IN SOLID MATRIX
WASTE PARTITION
TRANSURANICS PREPARATION
ENCAPSULATION
RECOVERED URANIUM & PLUTONIUM
FISSION PRODUCTS
LONG LIVED FISSION PRODUCTS PREPARATION
CURIUM
PREPARATION AND TERRESTRIAL STORAGE
TRANSPORT TO LAUNCH SITE
LAUNCH
SPACE DISPOSAL

FIGURE 6-1

EXTRATERRESTRIAL DISPOSAL PROCESS STEPS

The reference design chosen in the NASA study for the waste payload is shown in Figures 6-2 and 6-3. The waste is compacted and enclosed in coated tungsten spheres 3.3mm in diameter. These tiny spheres are mixed into a matrix of lithium hydride, copper, or aluminum for shielding and thermal conductivity. This large matrix is then compacted and enclosed in successive layers of coated tungsten, lithium hydride, and stainless steel. Design criteria have included the following:

- Radiation level of 1 Rem/hr or less at 1 meter
- Low temperatures throughout to avoid material degradation
- Ability to withstand launch fires, explosions, impacts
- Ability to withstand reentry temperatures, and pressures
- Ability to withstand surface impacts and burial

The design which has evolved for a payload to a solar system escape mission (Figures 6-2 and 6-3) has the following parameters:

- Outside diameter, 1.5 meters
- Outside diameter impact shell, 0.98 meters
- Total weight, 3,270 kg*
- Weight of transuranics, 113 kg
- Weight of fission products, 40 kg
- Weight of reentry shield, 4.5 kg
- Dose, 1 Rad per hour at 1 meter
- Thermal power, 9.2 KW

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*Weight capacity of the selected launch vehicle.

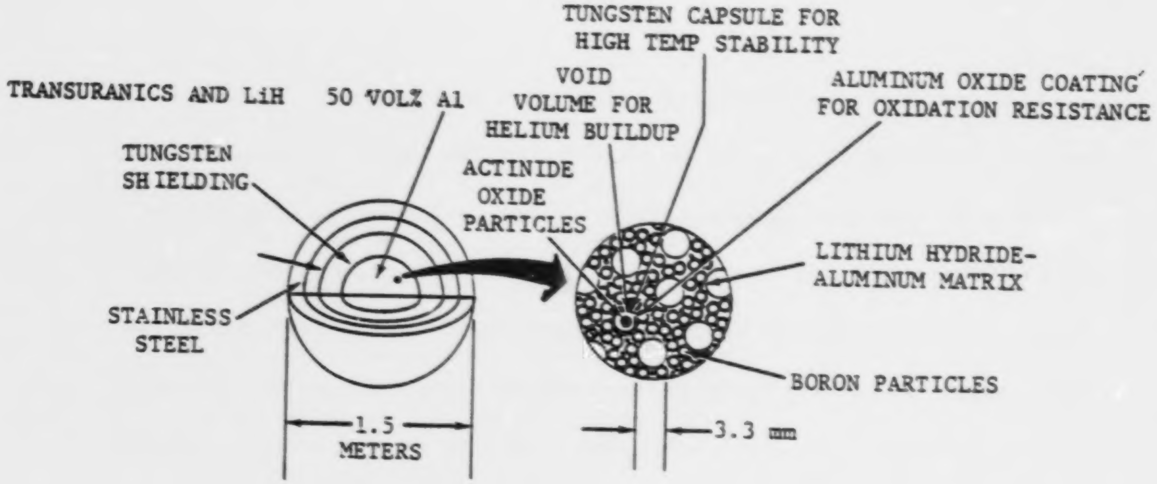
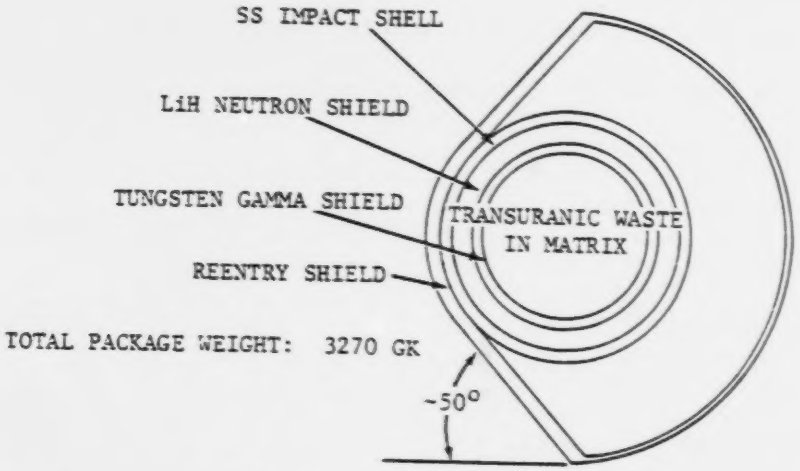

TUNGSTEN CAPSULE FOR HIGH TEMP STABILITY
TRANSURANICS AND LiH
50 VOL% Al
VOID VOLUME FOR HELIUM BUILDUP
ALUMINUM OXIDE COATING FOR OXIDATION RESISTANCE
TUNGSTEN SHIELDING
ACTINIDE OXIDE PARTICLES
LITHIUM HYDRIDE-ALUMINUM MATRIX
STAINLESS STEEL
BORON PARTICLES
1.5 METERS
3.3 mm

FIGURE 6-2
TRANSURANIC WASTE CAPSULE FOR SPACE DISPOSAL


SS IMPACT SHELL
LiH NEUTRON SHIELD
TUNGSTEN GAMMA SHIELD
TRANSURANIC WASTE IN MATRIX
REENTRY SHIELD
TOTAL PACKAGE WEIGHT: 3270 GK
~50°

Source: NASA TMX-2911, Lewis Research Center, Reference 1

FIGURE 6-3
REENTRY SHIELD AND TRANSURANIC DISPOSAL
PACKAGE FOR SOLAR ESCAPE DESTINATION

As will be evident in the subsequent discussion on safety, these precautionary measures provide a safety factor in the event of a launch accident. In addition, as shown in Figure 6-3, a reentry shield is required to protect the waste capsule in the event of accidental high velocity reentry into the atmosphere from space. The reentry shield provides for intact reentry of the waste package thus providing for enhanced potential for recovery. The rather sophisticated packaging of the waste minimizes the possibility of release on reentry impact and provides long-term containment in the event the waste package is not recovered.

The combined weights of the shielding, impact shells, and tungsten shields are nearly 2200 kg and the weight of the reentry shield is about 400 kg. Even modest reductions in shield weight would substantially improve the waste payload although obviously not on a one-for-one basis. The cost of encapsulation is of the order of a few percent of the cost of extraterrestrial waste disposal. It is clear therefore that cost is no barrier to efficient capsule design.

The features of the design which have been developed analytically or experimentally are as follows:

- Not breached by pressures of 2400 atmosphere
- Not penetrated by aluminum fragments with speeds up to 500 feet/second
- Not damaged by short term fireballs
- Inner shell contains waste in five minute solid propellant fires (shield is lost)

- Survives vertical ballastic reentry at 11 km/sec
- May be breached by impact on hard granite but may not release waste (contained in tungsten protective layers)
- Outer shell will rupture if deeply buried in earth but waste will be contained by inner shell
- In the various accidents to be considered there may be deformations or loss of shielding which could increase radiation

Comparison of these features to the accident environments previously discussed shows the design to be qualitatively favorable.

The encapsulation processes are quite complex. The state-of-the-art for these processes is in an advanced stage of development from many years of experience in encapsulation of radioisotope heat sources. Further research and development would be required for the waste but no fundamental problems are anticipated. Plant facility designs for encapsulation would be similar to existing facilities for these operations.

The waste fraction presumed to be launched has the approximate composition given in Table 6-I. Other long-lived fission products may be included as previously mentioned. The thermal power and radioactivity of the actinides from different reactor types are given in Table 6-II.

There are many other options for composition of the partitioned encapsulated fraction that can be considered. The fraction finally selected will optimize the benefits, risks, and costs. The extensive

TABLE 6-I

CHARACTERISTICS OF WASTE FOR FINAL DISPOSAL

Material	Atoms/cc x 10^{21}	g/cc	Total g in Single Sphere
Li-6	11.2	0.1120	6,325
Li-7	13.8	0.1610	9,092
Cu	18.9	1.9900	112,376
O	4.06	0.1080	6,099
Al	13.5	0.6050	34,164
H	25.0	0.0420	2,360
Np-237	1.57	0.6180	34,898
Pu-238	0.0124	0.0049	276
Pu-239	0.0552	0.0219	1,237
Pu-240	0.0409	0.0163	920
Pu-241	0.00600	0.0024	135
Pu-242	0.00391	0.0016	88
Am-241	0.112	0.0448	2,530
Am-243	0.185	0.0746	4,213
Cm-244	0.0440	0.0178	1,005

Note: Sphere volume = 56.6 liters
source reference 2

Source: Battelle Pacific Northwest Laboratories, Reference 2.

TABLE 6-II

THERMAL POWER AND RADIOACTIVITY OF TRANSURANICS IN 10-YEAR-OLD WASTE

	LWR-U		LWR-Pu		HTGR		LMFBR-AI		LMFBR-GE	
	Thermal Power[(a)]	Radio-activity[(b)]	Thermal Power[(a)]	Radio-activity[(b)]	Thermal Power[(a)]	Radio-activity[(b)]	Thermal Power[(a)]	Radio activity[(b)]	Thermal Power[(a)]	Radio activity[(b)]
Total Actinides Less U	69.9	2,350	1,230	36,900	617	25,000	169	7,140	141	5,530
Curium	60.4	1,727	1,144	32,617	36.7	1,051	41.3	959.1	57.5	1,633
Percent of Total in Curium	85	73	93	89	6	4	24	13	41	30

(a) Thermal power is in watts/MT of U + Th

(b) Radioactivity is in curies/KT of U + Th

Source: Battelle Pacific Northwest Laboratories,
High-Level Radioactive Waste Management Alternatives,
Section 8, May 1974

analysis required to determine the most favorable mix has not yet been performed.

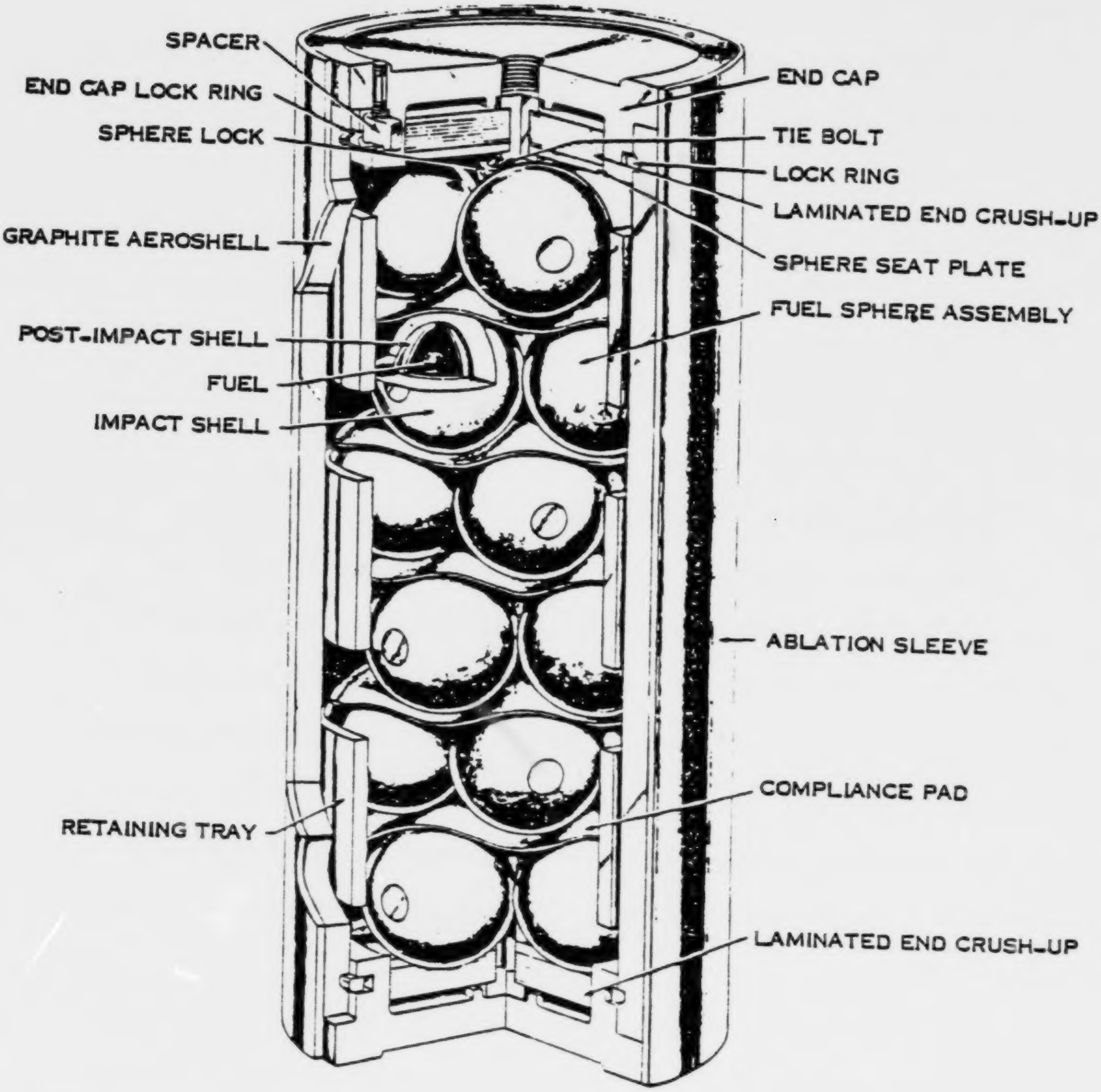
The actual payload, 113 kg of transuranics, requires over 3000 kg of protective encapsulation, or about 27-1/2 times the payload weight. Such a small payload margin, if decreased by further design and development refinement, could significantly increase the program costs. However, the conservation assumptions of the studies performed and the many potential options in design or choice of protective devices, shielding, and launch operations make it more likely that higher payloads could ultimately be achieved.

Alternative Waste Capsule Designs

There are a variety of waste capsule design approaches. One such approach is illustrated in Figure 6-4 which shows the reentry protection and encapsulation for a modern radioisotopic electric generator heat source.[5] The outer graphite cylinder provides the reentry protection and the inner graphite and metallic spheres provide the radioactive material containment. Extensive testing and analysis have shown this design to be safe for experimental flights on current launch vehicles.

Each heat source contains about 6 kg of $^{238}PuO_2$ and the assembly weighs approximately 20 kg. As many as four of these devices with 24 kg of fuel and 288,000 curies have been launched in a single flight.

Research and development for enhanced safety, reduced weight, and lower cost heat sources is continuing. One such concept is to


SPACER
END CAP LOCK RING
SPHERE LOCK
GRAPHITE AEROSHELL
POST-IMPACT SHELL
FUEL
IMPACT SHELL
RETAINING TRAY
END CAP
TIE BOLT
LOCK RING
LAMINATED END CRUSH-UP
SPHERE SEAT PLATE
FUEL SPHERE ASSEMBLY
ABLATION SLEEVE
COMPLIANCE PAD
LAMINATED END CRUSH-UP

Source: General Electric, Doc. No. 775054206, Reference 5.

FIGURE 6-4

MHW HEAT SOURCE

separate the radioactive material into a large number of small containers called "PADS."[6] This concept is illustrated in Figure 6-5. Some of the potential of total risk may be substantially reduced by more advanced capsule designs.

6.2.2 Launch Operations

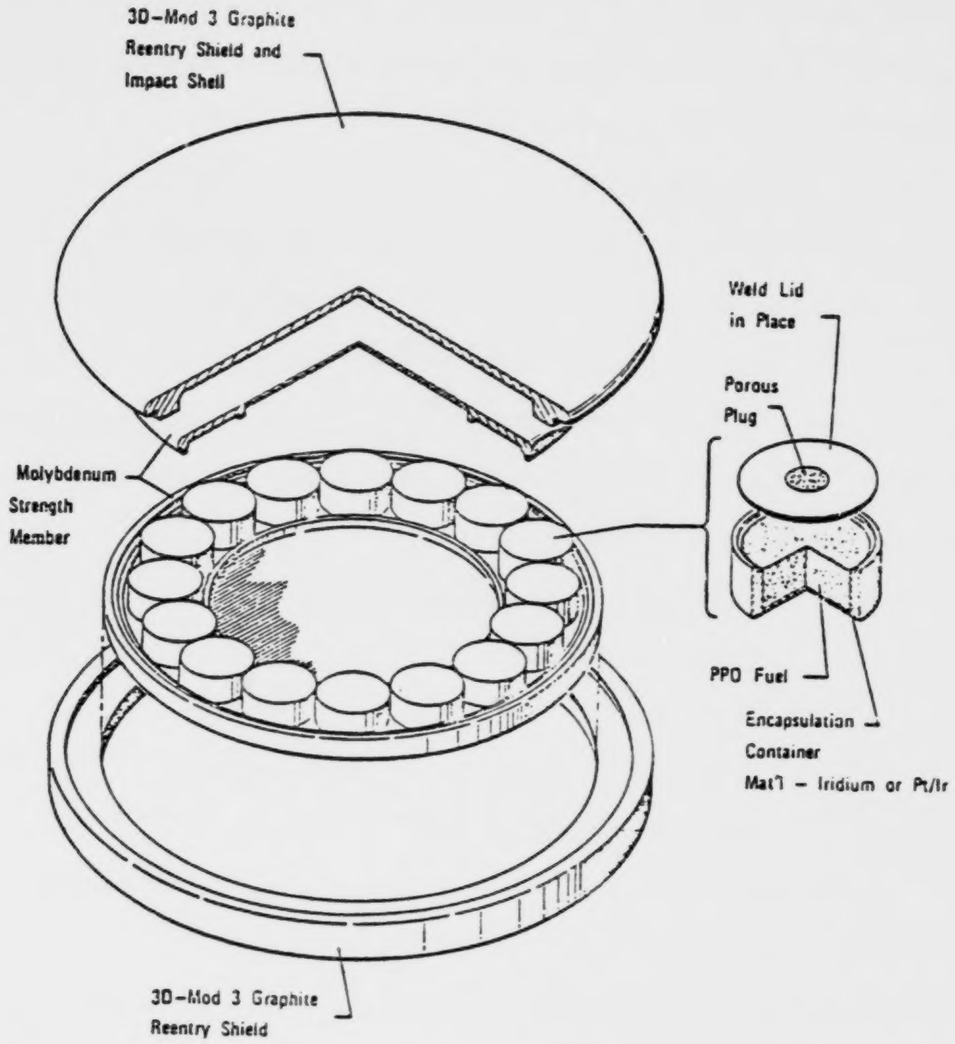
The space shuttle with its upper stages was selected in the NASA study because it performed the missions of interest at lowest cost. The costs for all other existing launch vehicles were appreciably higher. The launch vehicles considered are shown in Figure 6-6. The following list summarizes the cost of launch vehicles for high earth orbit or solar orbit destinations.[1]

Launch Costs, $/kg of Payload	
Titan III E/Centaur	4920
Saturn V	4590
Saturn V/Centaur	4390
Space Shuttle/Tugs	2940

All launched vehicles except the shuttle are scheduled to be phased out by the 1980s.

The destination studies in the NASA study were as follows:

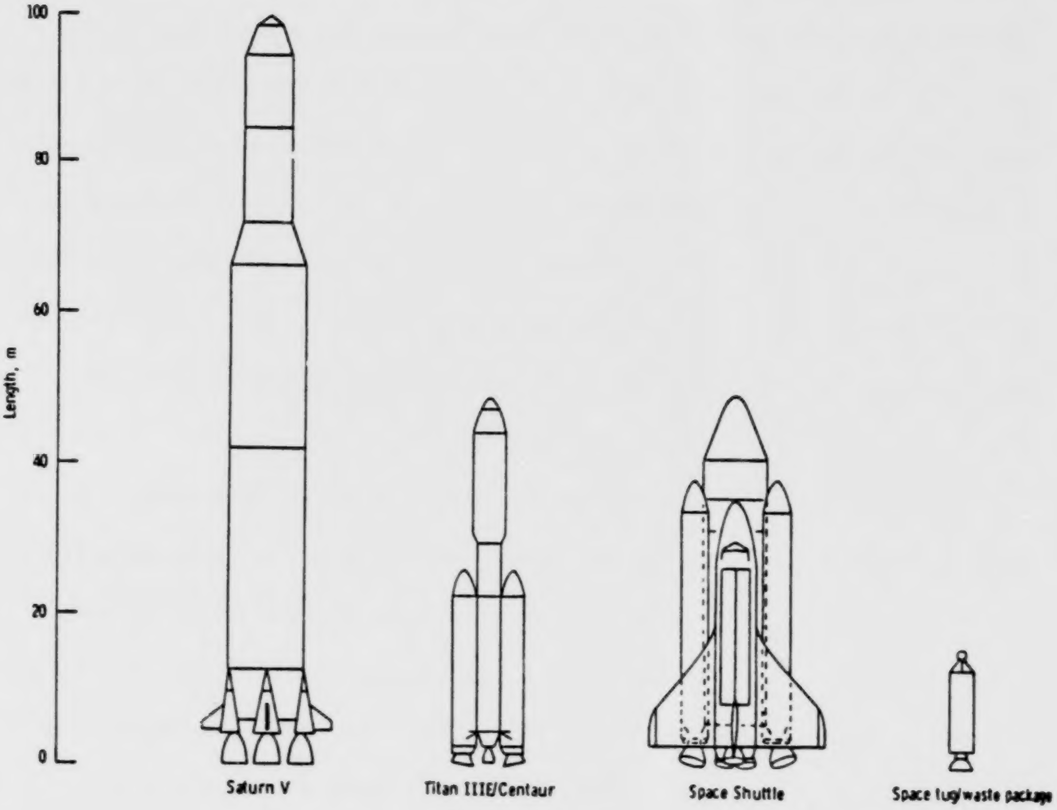
- High earth orbits
- Solar orbits
- Solar system escape
- Lunar impact or landing
- Planetary impacts


3D-Mod 3 Graphite
Reentry Shield and
Impact Shell
Weld Lid
in Place
Porous
Plug
Molybdenum
Strength
Member
PPO Fuel
Encapsulation
Container
Mat'l – Iridium or Pt/Ir
3D-Mod 3 Graphite
Reentry Shield

Source: BNWL-975, Battele Pacific Northwest Laboratories, Reference 6.

FIGURE 6-5

PAD CONFIGURATION



100
80
60
Length, m
40
20
0
Saturn V
Titan IIIE/Centaur
Space Shuttle
Space tug/waste package

Source: NASA TMX-2911, Lewis Research Center, Reference 1.

FIGURE 6-6

SPACE TRANSPORTATION SYSTEMS

Solar impacts are not possible without planetary swingbys because the velocity requirements are so high that present launch vehicles cannot provide the necessary boost. Planetary swingbys pose the possibility of contaminating their surfaces in violation of present international agreements.[2] For the same reason, planetary impacts are ruled out at the present time. High earth orbits and solar orbits are less attractive because there is no guarantee that the earth will not at some time recapture the waste, or portions of it, in the event that its encapsulation fails. High earth orbits and solar orbits are, however, more attractive than solar system escape on a cost basis in that space transportation cost could be reduced by a factor of four or five.

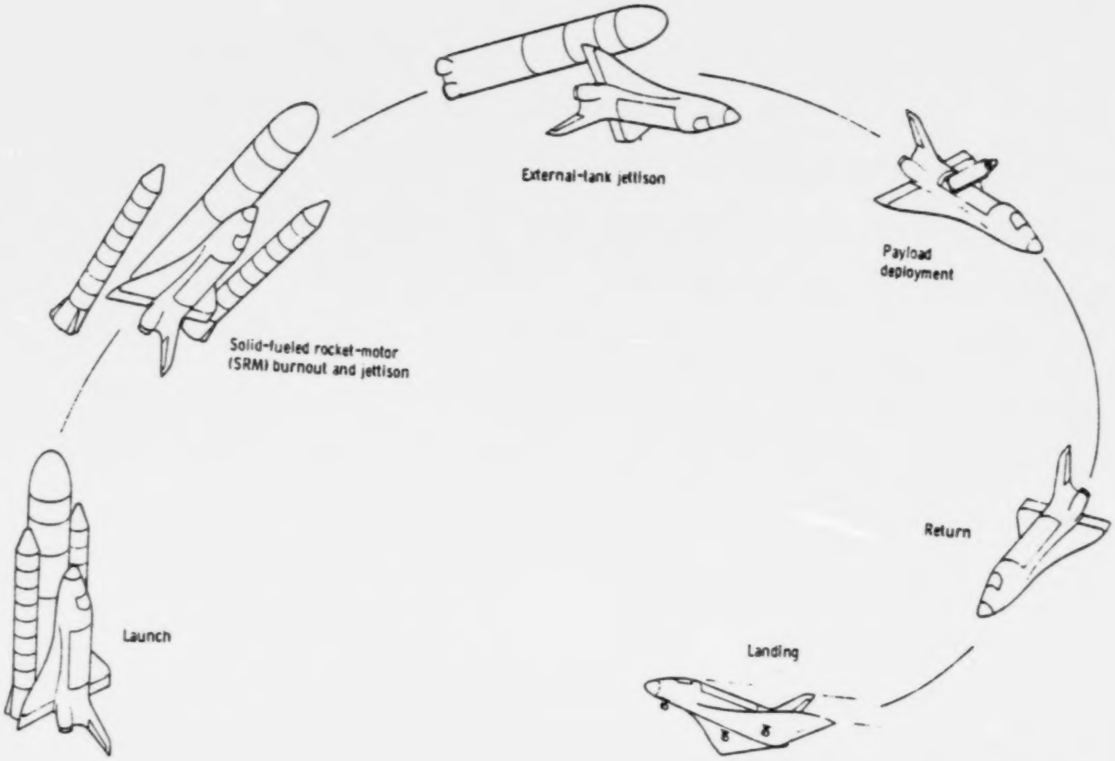
Lunar impacts or landings offer some potential advantages. Waste deposited on the moon could ultimately be recovered if that were to become desirable. The cost of lunar missions could also be attractive. The moon could be a useful staging point for finally launching the waste into deep space. However, current international agreements eliminate the lunar destination. The solar system escape destination is one which can be considered to permanently dispose of the waste for thousands to millions of years required and is the mission considered herein even though it is the most costly. It should be realized that for the very long time span that is contemplated for waste disposal, many advancements in launch vehicles, encapsulation, and other technologies may greatly reduce cost. Studies of extraterrestrial disposal are continuing and specific costs for today's technologies should not by themselves be the basis for permanently discarding the space option.

A typical space shuttle launch sequence is shown in Figure 6-7. Two such launches would be required for a solar system escape mission. In one launch, the payload would be an expendable tug upper stage with the waste capsule, and in the other, a reusable tug. The two tugs would rendezvous in high earth orbit and fire successively, accelerating the payload to escape velocity. Such missions are expected to be routine by the 1980s.

There is at least a daily launch opportunity for the solar escape mission. It may be targeted to miss planets without difficulty. The waste will escape the solar system in about twenty years. The number of shuttle launches per year required to dispose of the ten-year-old waste is shown in Figure 6-8. Depending on the composition of the waste, 100 to 250 launches per year would be required by the year 2000. If the launches are made from the existing launch facility (Kennedy Center) together with the normal anticipated space program, a modest expansion of launch facilities would be required. If other launch sites are to be considered, substantial expense would be involved in creating a new facility.

6.2.3 Technical Feasibility

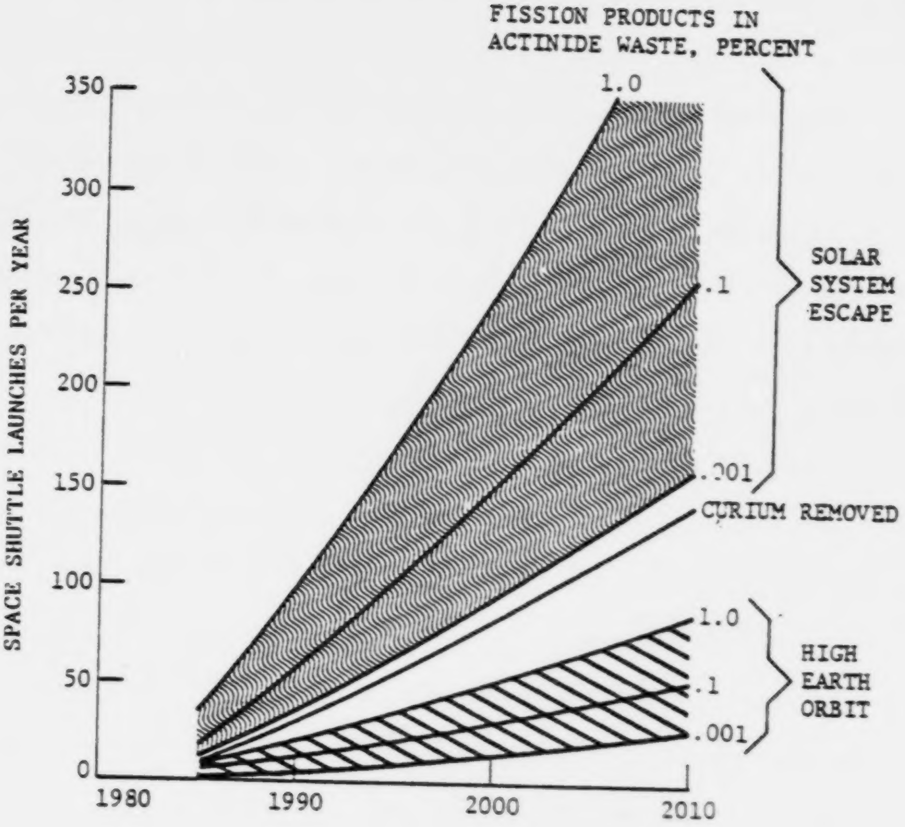
The entire technology of extraterrestrial waste disposal is in the conceptual stage with the exception of the space shuttle, which is in its development phase. However, the processes of partitioning


External-tank jettison
Payload deployment
Solid-fueled rocket-motor (SRM) burnout and jettison
Return
Launch
Landing

Source: NASA TMX-2911, Lewis Research Center, Reference 1.

FIGURE 6-7

SPACE SHUTTLE LAUNCH-TO-LANDING SEQUENCE



Source: NASATMX-2911, Lewis Research Center, Reference 1.

FIGURE 6-8
NUMBER OF SPACE SHUTTLE LAUNCHES REQUIRED PER YEAR FOR DISPOSAL OF ONLY ACTINIDES INTO HIGH EARTH ORBIT OR BY SOLAR SYSTEM ESCAPE. PRIOR 10-YEAR EARTH STORAGE.

and encapsulation are similar to those for fuel reprocessing and heat source encapsulation and therefore would benefit from a considerable experience background. The space launch operations are being developed as a part of the nation's current space program. The launching of radioactive materials has been commonplace in the past decade. As will be discussed subsequently, the optimization of launch operations and vehicle and capsule designs can substantially enhance safety.

It is evident that a research and development program of substantial scope and cost for adoption of the space program to radioactive waste disposal would be required to establish the final practicality even though it can be considered to be technologically feasible. Consideration of the magnitude of effort required to complete such a program would indicate a likely time span of around twenty years to maturity, although this could probably be shortened if a crash program were to be undertaken.

6.3 Environmental and Health Considerations

The environmental issues which concern extraterrestrial disposal of radioactive waste can be divided into two parts: those due to normal operations, and those due to abnormal events such as accidents or unplanned events.

6.3.1 Normal Operations

Normal waste extraterretrial operations include partitioning of waste and encapsulation, terrestrial transport, and space transport.

6.3.1.1 Partitioning and Encapsulation. Partitioning requires the construction and operation of plants similar in size and nature to the presently constructed fuel reprocessing plant, but differing in the detailed processes that will be used. These processes will depend on the ultimate choice of the fractions to be separated, the chemical composition chosen for the products, and the types of waste to be processed. Similarity to reprocessing plants leads to the conclusion that some chemical and radioactive material releases would be expected. Some thermal pollution of local water sources is likely and plant construction and land use will intrude on the local environment. These factors have been considered for a typical reprocessing plant in a Draft Environmental Statement and found to have minimal adverse effects on local environments or populations.[7]

Encapsulation of the waste involves an operation on a considerably smaller scale than fuel reprocessing or a waste partitioning operation. Accordingly, during normal operations, no significant releases would be anticipated and no significant environmental intrusions would be expected.

6.3.1.2 Terrestrial Transportation. Presumably, terrestrial transportation would conform to the Federal Code Part I 10CFR7, or its successor, which prescribes the normal and accident provisions for protection and transportation of radioactive materials.

Occupational and general public radiation exposure could potentially be higher than that for other disposal concepts in that both partitioned waste for extraterrestrial disposal and residual waste (fission products) for an alternative disposal require transport to the disposal or launch site. In the event that a remote island launch site were to be constructed, sea transport would also be involved (see Section 7.0, Seabed Disposal). No significant effect on the environment would be expected.

6.3.1.3 Space Transportation. Space shuttle launches at the Kennedy launch site are expected to approach 50 to 100 a year soon after the year 2000 and impose some safety hazard even if no radioactivity is released. Nuclear waste disposal missions could increase the frequency of launch by factors of 2 to 4 in the next few decades. Environmental studies by NASA[8] have identified several potential effects which, based on the current traffic models, are thought to be of minimal significance. These will, however, be more significant if the number of flights is substantially increased for radioactive waste disposal. These effects include noise and sonic boom, acidic rain, slight reduction in upper atmosphere ion concentrations, and the common local community interactions.

The environmental effects on the upper reaches of the atmosphere depend on the type of vehicle employed. The chemical effluents can cause reduction in the local ion concentration in the ionosphere, thus affecting radiowave propagation. A special type of acidic rain can occur from the propellant emission of hydrogen chlorine. The ozone depletion contribution for 100 launches is around 0.33 percent per year.[3]

The possibility of acidic rain, toxic emissions, launch noise, and sonic booms would make a remote island site more acceptable than an established launch area such as the Kennedy Space Center. The requirement for a new site will depend upon the number of flights as affected by the nuclear waste mix, the form chosen for disposal, and the results of future impact assessments.

The annual energy requirements for materials and propellants for one hundred space shuttle flights per year have been estimated to require 4×10^{13} kilojoules. This represents about 2.8 percent of the electric power to be generated in the year 2000, assuming an installed capacity of 638 GWe at an availability factor of 70 percent.

Increasing the launch role will increase the magnitude of these effects and also introduce the need for additional site development with some modest construction impact. Additional study would therefore be required to assess these factors. No radioactive releases would occur under normal launch operations.

6.3.2 Abnormal Events

Abnormal events, or accidents, have some potential of occurring at each stage of the waste handling process; partitioning and fractionation of waste, encapsulation of waste, transportation of waste and launch, and space transport of waste. A detailed assessment of the risk and consequences of extraterrestrial disposal has not been performed. Partitioning, fractionation, encapsulation, and transport of waste are not expected to be greatly different from operations that are currently performed in the nuclear industry and U.S. space programs. To the extent that these current operations are presently acceptable or will be acceptable for other disposal alternatives, it can be assumed that similar operations for extraterrestrial disposal can be made equally acceptable. The major difference between extraterrestrial disposal and other alternative concepts is the launch and space transport stages. This phase of the extraterrestrial disposal is of particular concern because of the potential consequences of failure. The balance is the rather complete removal of the waste and the corresponding elimination of risk to future generations.

6.3.2.1 Launch Vehicle Accidents. The typical launch accidents are summarized in Table 6-III together with a listing of the resultant events and requirements.

Accident evaluation is commonly divided into four phases:[9]

Phase 0 - Prelaunch

Phase 1 - Ignition until the impact point clears the launch area

Phase 2 - Ascent to parking orbit

Phase 3 - Parking orbit to escape

Prelaunch and launch area (Phase 0 and 1) could involve the following:

- Catastrophic explosion and fire
- System failure while the vehicle is near the launch pad

TABLE 6-III

TYPICAL LAUNCH ACCIDENTS

Launch Phase	Accident	Environment	Requirements	Comments
Phase 0	Mishandling of payload or propellant	Explosion, propellant fireball, impact on pad, liquid residual fire, solid propellant fire, impact of fragments	Comprehensive, rigid ground procedures with checks and controls	Very low probability
Phase 1	Explosion and fire, vehicle tumbling and impact	Similar to Phase 0	Capsule must not be breached by overpressure, light temperature, solid propellant fire, shrapnel or debris and burial	Source can be limited by integrated design of containment and launch vehicle and by operating procedures
Phase 2	Failure to reach earth orbit	Ballistic reentry, land or water impact	Capsule must withstand reentry temperature and pressure loads, impact loads, submergence, burial	Low probability with shuttle reentry capability.
Phase 3	Failure to reach escape velocity	Extended solar or elliptical earth orbit, possible reentry	Capsule inert in space environment, reentry similar to Phase 2	Possible recovery by tug in some modes

- Guidance and control system failure causing the vehicle to strike the ground resulting in explosion and fire

Ascent and parking orbit to escape accidents (Phases 2 and 3) could involve the following:

- Explosions and fire at high altitudes
- System failure resulting in short-lived orbit or powered reentry
- Maneuver or docking accidents with reentry

Present day and past launch vehicles were usually designed for unmanned operation and incorporated little or no redundancy. The Saturn V used for the Apollo missions and some versions of the Atlas and Titan vehicles used in the earlier manned missions employed limited redundance. In general, the liquid propellant vehicles have experienced a mission success reliability in the range of 88 to 100 percent with a median of 94 percent. Thus, six missions out of 100 failed to achieve the objective, but not all failures would result in loss of the waste. A recent Titan Centaur launch was calculated to have the following vehicle loss probabilities:[5]

Phase 0 - 0.0043

Phase 1 - 0.000712

Phase 2 - 0.0355 (with 0.025 land impacts)

Phase 3 - 0.01925

Total - 0.059762

Thus, six out of 100 such launches would be expected to experience vehicle loss whereas the overall vehicle mission reliability is estimated to be less than 90 percent. A vehicle loss is not normally expected to release radioactive material since the waste form and container can be designed to withstand extreme environmental conditions. Also, an intact waste container may be recovered.

6.3.2.2 Radioactive Waste Releases. Radioactive material could be released from accidents during every phase of the launch. Assuming that the future containment technology and chemical forms of the waste are at least equal to, and possibly superior to, the technology used for $^{238}PuO_2$ radioisotope heat sources utilized in present space programs,prelaunch accidents, i.e., prior to placement on a fuel launch vehicle, are unlikely to result in radioactive material releases. The most likely release areas would be the launch area during Phase 1 accidents and land or sea impact on a worldwide basis during Phase 2 and 3 accidents. The potential impact areas would be more closely defined in the event of Phase 2 accidents through flight path selection. The rather sophisticated designs for radioisotope heat sources are unlikely to burn up on reentry and result in atmospheric releases. Less sophisticated, though more economical designs may be more subject to release of radioactive waste in the atmosphere in the event of an accident.

As noted previously, containment capsules can be designed to withstand hostile environments. Therefore, any accident of sufficent severity to breach a modern container during Phase 1 accidents will necessarily involve explosion and fire. In many cases, and particularly in the case of the shuttle with its large inventory of hydrogen and oxygen and solid propellants, the fire will be of such intensity that a portion of the waste released will be vaporized. The radiological release during this phase will, in all probability, be vapor and larger particles near the launch site.

Phase 2 and 3 accidents may produce impacts on land or water or, though less likely, upper atmosphere releases. Modern containers will not be breached by most impacts. However, some probability exists that impact on hard rock may cause a potential release. The release in the event of hard rock impacts will consist of a respirable fraction and larger particles. The respirable fraction will become airborne and will then settle out in accordance with dispersion mechanisms. If disturbed by natural or artificial events, the particles may become partially resuspended and be redeposited.

In the event of a reentry of the waste capsule and water impact, the time and rate of release will be dependent upon the damage sustained by the capsule (fire and/or explosion of vehicle, reentry heating, impact damage, hydrostatic pressure). If the waste capsule is buried in unconsolidated sediments of low thermal conductivity, the container may fail as a result of high temperatures.

The assessment of the risk associated with the space launching of radioactive materials is a complex and difficult task. Each potential failure mode of the mission must be examined and a probability determined. Each instigating failure will in turn have branching probabilities of events that may lead to a resulting accident of sufficient magnitude to lead to the release of radioactive materials.

A detailed analysis has not been conducted to predict the accident probabilities and the associated consequences for extraterrestrial waste disposal.

Although a detailed risk assessment has not been conducted, the analysis performed for the radioisotope thermoelectric generator designed for space applications provides insight to the risk and quantities of released material that might be expected. For the type of isotopic heat source shown in Figure 6-4, an analysis was conducted for a final safety analysis report which presented the probabilities for release of $^{238}PuO_2$ from affected fuel sphere assemblies (FSA).[5] In this analysis, the isotopic heat source contained twenty-four Fuel Sphere Assemblies with a total of 7×10^4 curies of Pu-238. The potential mission accidental release events and prompt fuel release probabilities are presented in Table 6-IV.[5] The predicted quantities of release and corresponding probabilities are given in Table 6-V. When an explosion and fire exist, the vaporized material will be lifted by the fireball and will have an effective height of release (H_{eff}) above the launch area as shown in Table 6-V. In the analysis for this radioisotope thermoelectric generator, the predicted probabilities of prompt release were small (on the order of 10^{-5} to 10^{-8}) and the predicted quantities of release were also a small fraction (10^{-4}) of the total inventory.

The analysis conducted for the RTG is not directly applicable to radioactive waste disposal. In particular, the ratio of total weight

TABLE 6-IV

MISSION POTENTIAL FUEL RELEASE EVENTS

Mission Phase	Initial Accident	Mechanism Causing Fuel Release	Location Affected	Mission Probability of Fuel Release
0 - Prelaunch	none	none	none	node
1 - Launch Area	A. Explosion and fire in Centaur	1. Spacecraft impacts on concrete launch pad side-on with RTG's hitting first	Launch Pad	
		a. No contact with burning UTP-3001		5.0×10^{-5}
		b. Contact with burning UTP-3001		5.0×10^{-7}
		2. Spacecraft impacts on sand near launch pad side-on with RTG's hitting first	Launch Complex	8.3×10^{-5}
	B. Tumbling vehicle-guidance/control malfunction	1. Centaur/SC impacts on concrete launch pad nose first	Launch Pad	
		a. No contact with burning UTP-3001		9.5×10^{-6}
		b. Contact with burning UTP-3001		1.6×10^{-7}
		2. Centaur/SC impacts on concrete launch pad side-on with RTG's hitting first	Launch Pad	
		a. No contact with burning UTP-3001		1.3×10^{-5}
		b. Contact with burning UTP-3001		2.1×10^{-7}
		3. Centaur/SC impacts on sand near launch pad side-on with RTG's hitting first	Launch Complex	2.2×10^{-7}
2 - Ascent	Spacecraft ballistic re-entry due to launch vehicle malfunction	HSA impacts on rock following re-entry	Ground Track	
		a. High velocity impact		2.2×10^{-7}
		b. Low velocity impact		5.1×10^{-7}
3 - Orbit	Launch vehicle malfunction resulting in prompt re-entry or orbit decay			
	1. Multiple skip re-entry	HSA impacts on rock following re-entry	28° N to 28° S	
		a. High velocity impact		6.3×10^{-10}
		b. Low velocity impact		1.5×10^{-9}
	2. All other re-entries	HSA impacts on rock following re-entry	28° N to 28° S	
		a. High velocity impact		1.5×10^{-5}
		b. Low velocity impact		3.5×10^{-5}

Source: General Electric, Doc. No. 77SOS4206, Reference 5

TABLE 6-V

MISSION PROMPT SOURCE TERM SUMMARY

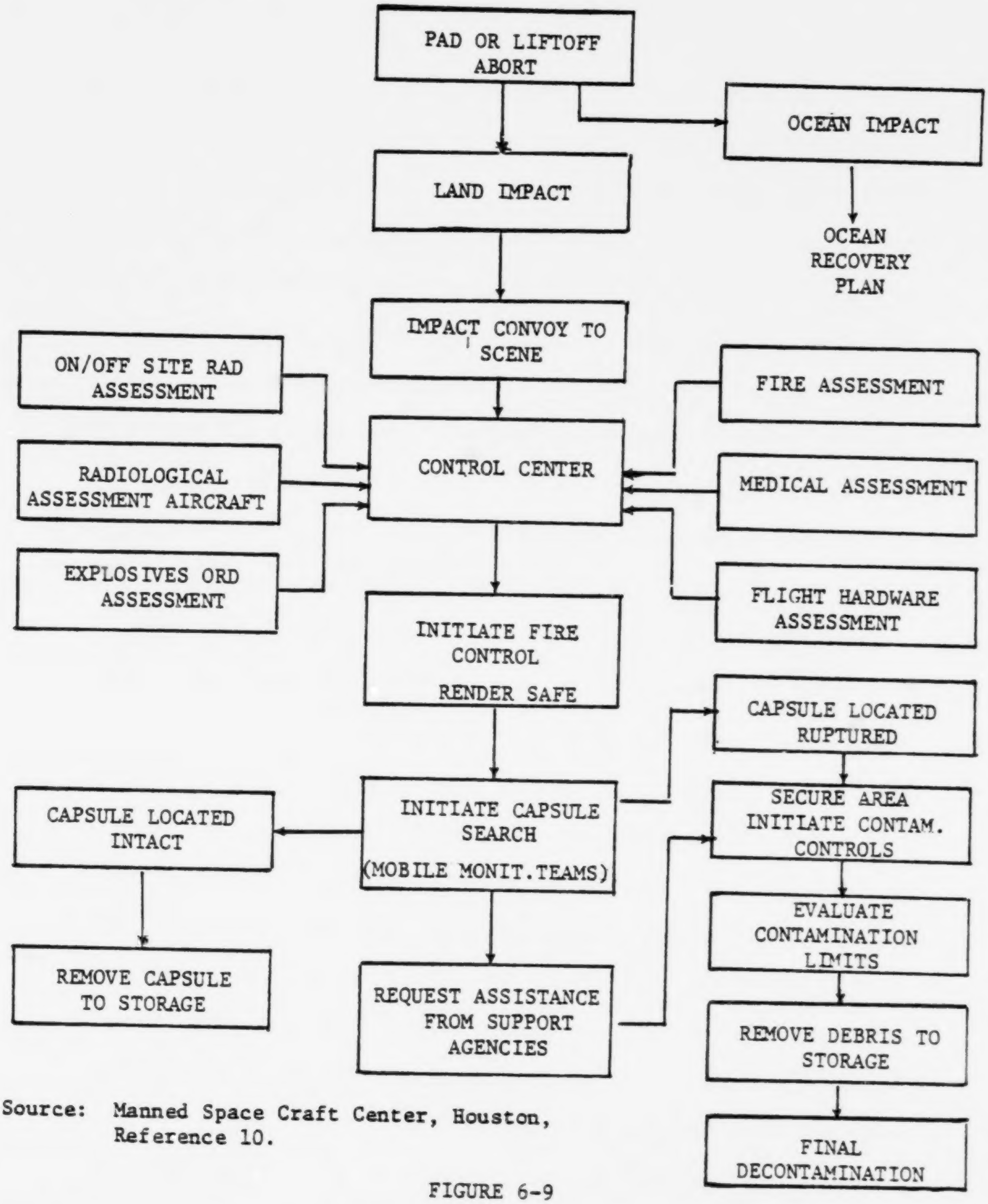
Mission Phase	Region Affected	No. of FSAs	Probability	Location	H_{eff} Meters	Amount Released mCi			Max. No. FSAs	Probability	Amount Released mCi		
						<4μ	Vapor	Total			<4μ	Vapor	Total
Launch Area (1)	Launch Pad	1	1.6 (-6)	air	260-3970	-	387	387	12	6.7 (-26)	-	3012	3012
		1	4.5 (-7)	air	105-1595	-	387	387	12	9.0 (-21)	-	3012	3012
		1	3.1 (-6)	air	260-3970	-	183	183					
		1	8.9 (-7)	air	105-1595	-	183	183					
		1	1.3 (-7)	air	260-3970	-	387	387	4	1.0 (-12)	-	936	936
		1	1.5 (-8)	air	105-1595	-	387	387	4	4.1 (-12)	-	936	936
		1	4.0 (-7)	air	260-3970	-	183	183					
		1	4.6 (-8)	air	105-1595	-	183	183					
		3	1.5 (-7)	surface	1	3.2	-	549	4	9.1 (-6)	5.1	-	936
		3	4.5 (-7)	surface	1	4.0	-	753					
		11	2.0 (-6)	surface	1	14	-	2620	12	5.6 (-5)	16	-	3012
		11	4.0 (-6)	surface	1	15	-	2830					
	Launch Complex	6	8.3 (-5)	buried	-	6.5	-	1098	-	-	-	-	
Ascent (2)	Ground Track	2	3.3 (-7)	surface	1	2.7	-	366	6	7.8 (-14)	8.1	-	1098
		5	3.3 (-7)	surface	1	6.8	-	915	15	7.8 (-14)	20.3	-	2745
		2	1.5 (-6)	surface	1	2.2	-	366	6	3.6 (-13)	6.5	-	1098
Orbit (3)	28° N-28° S	3	2.2 (-5)	surface	1	3.2	-	549	9	7.3 (-11)	9.7	-	1647
		2	9.3 (-10)	surface	1	2.7	-	366	6	3.0 (-15)	8.1	-	1098
		5	9.3 (-10)	surface	1	6.8	-	915	15	3.0 (-15)	20.3	-	2745
		2	4.4 (-9)	surface	1	2.2	-	366	6	1.4 (-14)	6.5	-	1098

Source: General Electric, Doc. No. 77SDS4206, Reference 5 to weight of waste may be significant if large quantities must be transported. The analysis performed for the RTG does indicate, however, that the risk of prompt release of waste in extraterrestrial disposal can be reduced to low values. To determine whether this will lead to an excessive number of required flights, whether the risk is acceptable, or whether the costs are unreasonable will require further system design and evaluation.

Regardless of whether the waste capsule survives the accidental event, it must be assumed to fail prior to the radioactive decay elimination of the long-lived radioisotopes. Recovery of accidentally released waste capsules is therefore an important aspect of extraterrestrial disposal.

6.3.3 Recovery and Contingency Planning

Safety during space launches has been enhanced by the use of operational procedures that have been developed to counter accidents that may occur and procedures developed to isolate and recover radioactive material. Figure 6-9 partially illustrates the sequences of actions that would be undertaken in the event of a Phase 1 accident. Should an accident occur, immediate measures, as indicated, are undertaken to ensure the safety of the launch personnel and the public, the protection of the environment, and to expeditiously recover the radioactive material. These practices have been refined and improved and are presently operational.[10]


PAD OR LIFTOFF ABORT
OCEAN IMPACT
LAND IMPACT
OCEAN RECOVERY PLAN
IMPACT CONVOY TO SCENE
ON/OFF SITE RAD ASSESSMENT
FIRE ASSESSMENT
RADIOLOGICAL ASSESSMENT AIRCRAFT
CONTROL CENTER
MEDICAL ASSESSMENT
EXPLOSIVES ORD ASSESSMENT
FLIGHT HARDWARE ASSESSMENT
INITIATE FIRE CONTROL RENDER SAFE
CAPSULE LOCATED RUPTURED
CAPSULE LOCATED INTACT
INITIATE CAPSULE SEARCH (MOBILE MONIT.TEAMS)
SECURE AREA INITIATE CONTAM. CONTROLS
EVALUATE CONTAMINATION LIMITS
REMOVE CAPSULE TO STORAGE
REQUEST ASSISTANCE FROM SUPPORT AGENCIES
REMOVE DEBRIS TO STORAGE
FINAL DECONTAMINATION

Source: Manned Space Craft Center, Houston, Reference 10.

FIGURE 6-9

RADIOLOGICAL RECOVERY SEQUENCE

If an accident should occur during Phase 2 or 3, return to the earth would be on land or water remote from the launch site. In either case, there is a high probability that the capsule or capsules can be located through a combination of worldwide tracking during the descent, signal devices in the payload, and aircraft search and detection devices. Such aircraft are already available and have been used in several accident situations. Water recovery is possible and has been accomplished to a substantial depth, but not at all depths. The capsule design must therefore prevent the catastrophic release of radioactive material in the deep ocean.

Current detection and recovery capabilities are scaled to very rare occurrences of emergencies. It is likely that substantial enhancement of these operations would be necessary in the event that space waste disposal is used.

6.3.4 Shuttle, Waste Capsule Integration

There are several significant implications to be drawn from the information available. Safety is very strongly determined by integrated vehicle characteristics, encapsulation techniques, and operational activities. Experience has shown that the proper combination of these can substantially reduce the risk of space flight operations. Research and development are continuing and further enhancement of safety can be expected in future missions.

The space shuttle and tug combinations represent the most advanced state-of-the-art of launch vehicle design presently known.

It is anticipated that catastrophic failures would be substantially less probable than for former launch vehicles. The combination of the shuttle's ability to recover safely from many previous accident situations, its redundant systems, and the presence of a pilot with capability to take remedial actions will contribute to a reduced probability of accidental releases. The ability of the tugs to rendezvous and recover waste from aborted missions will also contribute to the safety of space operations.

A wide variety of encapsulation and system approaches is possible. For example, it is possible to consider such options as launching only small amounts of waste at one time as "piggy back" payloads for shuttles that are not fully loaded and collecting them at a space depot. While the practically of such concepts remains to be determined, there are many options to optimize safety and minimize risk.

6.3.5 Radiological Considerations

Disposal in space of a fraction of the nuclear waste may affect the ecosystem during normal operations and, in the event of accidents, may result in the release of radioactive materials to the environment. The primary concern in extraterrestrial waste disposal is accidents during the launch phases.

The steps to compute radioactive waste release consequences are as follows:

1. Determine the probability of the accident
2. Determine the source terms and their probability

3. Predict the movement or dispersion of released material through the environment by atmospheric or aquatic dispersion processes

4. Determine the probable number of people exposed and the probable doses received

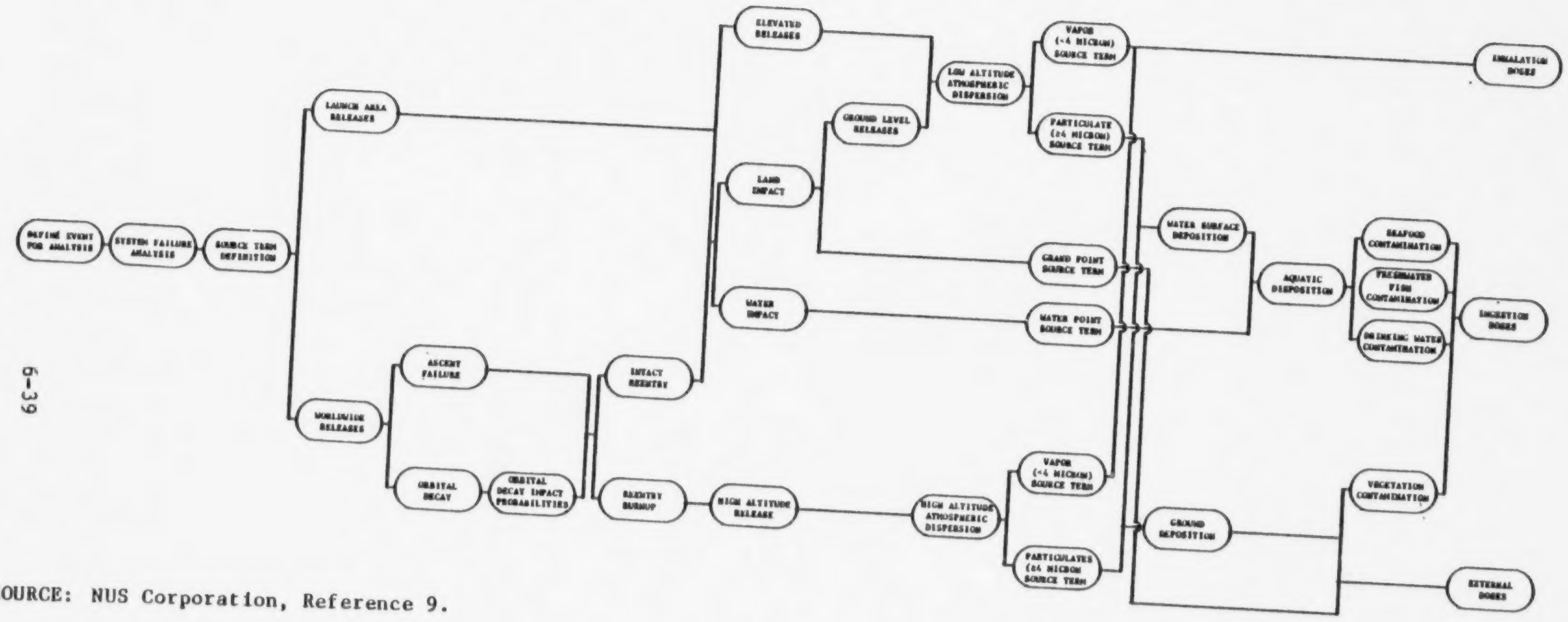
A generalized diagram for risk analysis in a space vehicle launch sequence is shown in Figure 6-10. Such an analysis has not been conducted for the space disposal of radioactive waste. However, recent studies have been conducted by Battelle Columbus for the National Aeronautics and Space Administration (NASA) based on worst case analysis.[3]

The Battelle study assumed a 5500 kg nuclear waste payload in a calcine powder waste form. The study considered five options of waste mix and five types of abnormal events or accidents.

The waste mixes considered were as follows:

- The fuel is leached from its clad and the entire dissolved solution is solidified and shipped into space;
- 99.5 percent of the uranium is recovered from the dissolver solution. The remaining dissolver solution is solidified and sent into space;
- 99.5 percent of the uranium and plutonium is recovered from the dissolver solution. The remaining dissolver solution is solidified and sent into space;
- 0.1 percent of the uranium and plutonium, the balance of the actinides and all the rare earths except cerium are solidified and sent into space;
- 99.5 percent of the uranium and plutonium and a minimum of 94 percent of the technetium are recovered from the dissolver solution. Only the technetium is sent into space.

The abnormal events considered were as follows:


DEFINE EVENT FOR ANALYSIS
SYSTEM FAILURE ANALYSIS
SOURCE TERM DEFINITION
LAUNCH AREA RELEASES
WORLDWIDE RELEASES
ASCENT FAILURE
ORBITAL DECAY
ORBITAL DECAY IMPACT PROBABILITIES
INTACT REENTRY
REENTRY BURNUP
HIGH ALTITUDE RELEASE
ELEVATED RELEASES
LAND IMPACT
WATER IMPACT
GROUND LEVEL RELEASES
LOW ALTITUDE ATMOSPHERIC DISPERSION
HIGH ALTITUDE ATMOSPHERIC DISPERSION
VAPOR (<4 MICRON) SOURCE TERM
PARTICULATE (≥4 MICRON) SOURCE TERM
GRAND POINT SOURCE TERM
WATER POINT SOURCE TERM
VAPOR (<4 MICRON) SOURCE TERM
PARTICULATES (≥4 MICRON SOURCE TERM
WATER SURFACE DEPOSITION
GROUND DEPOSITION
AQUATIC DEPOSITION
SEAFOOD CONTAMINATION
FRESHWATER FISH CONTAMINATION
DRINKING WATER CONTAMINATION
VEGETATION CONTAMINATION
INHALATION DOSES
INGESTION DOSES
EXTERNAL DOSES

SOURCE: NUS Corporation, Reference 9.

FIGURE 6-10
GENERALIZED FLOW DIAGRAM FOR RISK ANALYSES

- On- or near-pad catastrophic launch vehicle explosion and fire (major impact to lower atmosphere);
- Launch vehicle explosion and fire at high altitude or reentry and burnup of the nuclear waste payload from orbit (major impact to upper atmosphere--followed by chronic impact to lower atmosphere);
- Water impact resulting from launch vehicle failure or intact reentry of the nuclear waste payload from orbit (major impact to ocean or fresh water);
- Land impact resulting from launch vehicle failure or intact reentry of the nuclear waste payload (major impact to land).

The entire inventory of 5500 kg was assumed to be released in the worst case analysis.

The conclusion of the Battelle study is as summarized below:

> "For the five types of events considered here, a catastrophic on- or near-pad launch vehicle failure at KSC, resulting in the rupture and release of the radioactive waste payload, is considered very serious. A high altitude burnup is considered serious, with other events following in severity (sea, land, and space lunar type accidents). The assessment of effects to man and ecosystems as a result of these events is extremely difficult. However, order of magnitude projections can be made.
>
> "In the case of the catastrophic on- or near-pad launch vehicle failure, assuming Mix No. 3, the local human population exposed to the cloud resulting from the fireball containing the radioactive material could receive an inhalation radiation dose exceeding 500 times background (background 0.1 rem/year). At downwind distances of 100 km, exposures could exceed 100 times background. Public radiation standards vary from 5 to 15 times background, depending upon the organ or part of the body exposed. If gravitational settling of radioactive particles were a predominant effect, the area downwind of the event would become severely contaminated, many life forms would be destroyed, and the land area would have to be isolated indefinitely.
>
> "An upper atmospheric burnup of the payload could result in similar effects, depending upon the particle size distribution of the radioactive material, and the longitude and latitude of the event. Chronic toxicity effects would be expected for the case of worldwide distribution of the material. The amount of strontium-90 which could be released by one accident involving

Mix No. 3 amounts to 40 percent of that released from all nuclear devices through 1962.

"For accidents at sea resulting in the release of radioactive material, exposure to man would primarily be by bioaccumulation of nuclides in aquatic food chains. Reduced productivity of aquatic organisms could limit food supplies to man.

"Effects caused by a nuclear waste package crashing on land, in a populated area, followed by dispersal of the waste (calcine powder) could be significant. If the waste material is characterized by a fine particle distribution, then the chance for resuspension in the air becomes likely, thus causing severe impacts to local human, plant and animal populations.

"Accidents in space, followed by radioactive releases are not expected to impact the earth's biosphere; however, contamination of orbital regions or other celestial bodies (especially the moon) could preclude the use of an orbit or as a future resource. Strong opposition would be expected from the scientific community, if it were likely to contaminate the moon or other planets by a waste disposal accident."

The worst case analysis is not, of course, representative of the expected consequences of an accident and can be considered improbable. The analysis does indicate, however, that a detailed risk and consequence study is required to assess the acceptability of extraterrestrial disposal. It is not possible at this time to specify allowable or critical dosage levels for the undetermined waste disposal fractions to be launched. Acute toxicity at dose levels in the range of 50 rems per year will result in some deaths. Chronic toxicity for dose levels of a few mrem per year may introduce genetic and fertility effects.

All of the dose magnitudes (except the short term airborne respiratory particles and the vaporized fraction) may be greatly affected by recovery and corrective measures. In the case of launch area accidents, recovery and corrective action can probably be completed in a few days. Outside the launch area the recovery time is probably a function of population density ranging from a few days to longer periods, depending upon the location of the waste capsule. Recovery will be important in reducing the hazard and the long-term risk to future populations.

In contrast to the individual launches of the past decade, space waste disposal will require many routine, repetitive space operations. By the year 2000, or shortly after, as many as several thousand radioactive payloads may be launched. It might be reasonably projected that the probability of a prompt release accident for a single flight could be in the range 10^{-5} to 10^{-7}. The corresponding long-term probability for several thousand launches could be in the range of 10^{-2} to 10^{-4}. It would therefore be anticipated that a small number of accidents would occur with some release.

A small number of accidents and the minor releases that would occur in a program in which vehicles, capsules, and operational procedures are optimized in the manner previously discussed would be highly unlikely to serious affect the ecology as a whole. Local impacts could be significant, however. System designs would therefore be required which would limit ecological impacts to inconvenience rather than injury. These system designs would, of course, be established in the context of the weighing of the overall long-term benefits to the public of permanent waste disposal against the degree of risk and cost.

6.4 Economic Impacts

The cost of space disposal estimated in reference 2 (and similarly in reference 1) in 1973-74 dollars is as follows:

	Cost, $/Kg waste
Partitioning	14,000
Encapsulation	4,700
Space launch	150,000
Total	168,700

The cost in mills per kw hr generated is 0.5 mills/kWh.

The cost in percent of cost of generated electricity (early 1970's) is 5 percent.

These costs do not include the cost of disposal of the separated wastes remaining on the earth. A more modern estimate is required to account for escalation of the past few years. It will then be instructive to identify and add the cost of disposal of the wastes left behind and finally to determine the incremental cost for space disposal so that its benefits may be weighed against cost.

6.4.1 Partitioning

For the purposes of this draft and pending an accepted number, an escalation update is assumed as follows:

$14,000 $/MT escalated 6 percent per year for 3 years

14,000 x 1.19 = 16,674 $/MT

6.4.2 Encapsulation

Encapsulation costs given in reference 2 are based on existing designs scaled to the capacity required to meet waste disposal needs and are broken down approximately as follows:

Labor, materials and labor related	$2,000/kg
Construction (capital)	$2,700/kg
Escalation of labor and materials by six percent per year and capital costs by 12 percent per year for three years	$6,200/kg

Because this cost is small relative to the other costs involved, high precision is not needed.

6.4.3 Space Launch Costs

Shuttle and tug costs used in the references are broken down as follows:

Each shuttle flight	$10.5 M
Each reusable tug flight	$ 1.75 M
Each expendable tug	$ 5.5 M

One complete payload launch $28.25 M plus $.5M per flight for new launch facilities or total = $28.75M.

Actual operational costs are not known at this time. The most recent published estimates available have been given in a September 1976 statement by NASA Associate Administrator, J. F. Yardley, to the U.S. House of Representatives Committee on Science and Technology as follows:

Cost of each shuttle flight in 1975 $, Million

Commercial and Foreign	19.0 - 20.9
Other U.S. Government	16.1 - 18.0
DOD	12.7 - 14.1

Assuming that waste disposal can be considered to be a government activity, a cost of $17M per flight is assumed.

The new cost per shuttle flight escalated to 1977 dollars is:

$17 \text{ x } (1.06)^2 = \$19.1\text{M}$

The growth ratio is $\frac{19.1}{10.5} = 1.82$.

Applying this same ratio to tug costs yields the following vehicle costs for one payload:

	$M (1977)
Two shuttles	38.2
One reusable tug	3.2
One expendable tug	10.0
Escalated facility costs	.7
Total	$ 52.1M
or 52.1/113 kg	$450,000/kg.

The new total cost breakdown is as follows:

Separation	17,000 $/kg
Encapsulation	6,000 $/kg
Vehicle	450,000 $/kg
Total	$483,000 $/kg

The new cost ratio is $\frac{482,000}{168,700} = 2.8$.

The cost per kWh estimated in references 1 and 2 was 0.5 mills/kWh. Assuming constant electric energy generation per kilogram of waste, then the cost per kWh is $0.5 \times 2.8 = 1.4$ mills/kWh.

The above is provided only as an estimate of cost. The cost of disposal of separated wastes remaining on earth must also be added. Although the cost is higher than other disposal methods, it is not necessarily the limiting factor. The consequences of potential accidents is the more important consideration.

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7.0 SEABED DISPOSAL

In this section, the emplacement of radioactive wastes in deep-sea sediments is discussed relative to the technical feasibility and environmental acceptability of seabed disposal.* The technical feasibility of the concept depends upon demonstrating that seabed disposal can contain radioactive waste long enough for it to decay to innocuous levels. The time required for some long-lived actinides and fission products to decay to innocuous levels is several million years, a time period for which long-range predictions are somewhat tenuous at best. The environmental acceptability must therefore be assessed as to the degree of long-term isolation and the potential radiological impacts of seabed disposal on the marine environment and to man. A discussion of the environmental impact assessment of seabed disposal will be made by dividing the high-level radioactive and transuranic contaminated wastes into distinctive components (actinides, select fission products, volatiles, etc.). The effectiveness of seabed disposal for each component can be compared and will help identify potential environmental problems.

Seabed disposal has been explored by several countries as a means of permanent disposal of high-level radioactive and transuranic wastes. Currently, there are no accepted international criteria or standards to guide individual national efforts. The International

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*Seabed disposal is the emplacement of waste within the seabed sediment or geologic formations in such a way as to ensure long-term containment. It is not to be confused with ocean-dumping.

Atomic Energy Agency (IAEA) has recently expanded its waste management programs to evaluate several proposed high-level waste disposal options including seabed disposal. However, waste management programs in the nations producing nuclear power are still in very early stages of development, and serious efforts by the IAEA to solve the waste problem on an international level are just beginning.[1] A series of three advisory group meetings have been held by the IAEA with the task of developing definitions and guidelines for seabed disposal.[2]

The public concern today over the radiological consequences of seabed disposal, in part, is based on past marine disposal practices of the U.S. and other industrial nations. Between 1946 and 1970, for example, the U.S. Atomic Energy Commission (AEC) licensed the disposal of more than 86,000 containers of low-level wastes (totaling 94,000 curies) into the Atlantic and Pacific Oceans. Britain disposed about 45,000 curies of low-level radioactive wastes into the Atlantic from 1951-1966.[1]

From a scientific point of view, it is very difficult to determine if damage has occurred or if a real hazard exists as a result of international radioactive waste disposal practices. In this regard, the U.S. has taken a leading role to protect the marine environment from pollution including disposal and dumping of radioactive wastes into the oceans.

Under the Marine Protection, Research and Sanctuaries Act of 1972, EPA was given authority to issue permits for disposal of low- and medium-level radioactive wastes into the ocean, but EPA has no similar control over high-level wastes. Congress would have to amend the Act, if the government decided to implement any form of sub-seabed disposal of high-level wastes.

The Nuclear Regulatory Commission (NRC) presently has jurisdiction over the licensing of radioactive waste repositories while the EPA has authority over the establishment of standards and regulations for the placement of radioactive waste into the ocean. The Department of Energy (DOE) is responsible under the National Environmental Policy Act (NEPA) for the environmental assessment of planned high-level waste disposal techniques, including seabed disposal. While specific criteria and standards for new regulations for waste management are still to be developed, recently established NRC goals include the following:[1]

- Isolation of radioactive waste from man and his environment for specific periods to assure public health and safety and preservation of environmental values;
- Reduction to as low a level as is reasonably achievable of
  - (a) the risk to public health both from chronic exposure associated with waste management operations and possible accidental releases of radioactive materials from waste storage, processing, handling, or disposal;
  - (b) long-term commitments such as land-use withdrawal, resource commitment, and surveillance requirements.

Thus, the ultimate evaluation of the potential DOE seabed disposal concept by the NRC and EPA will have to be made with an established set of technical, social, and environmental criteria and standards.

This study will discuss the present state of knowledge on seabed disposal and will assess the radiological impact to man and possible damage to the marine ecosystem from emplaced wastes. Seabed disposal, as defined in this study, is the controlled emplacement of radioactive waste in deep-sea sediments or rock formations under the ocean. The evaluation will carefully identify the transport processes by which radionuclides could migrate from the emplacement site through the metal canisters and the deep-sea sediment and the ocean column to the biosphere.

Physical and environmental barriers that may prevent migration of radionuclides exist. On the other hand, several mechanisms may act singularly or in combination to compromise the integrity of these barriers. Included among these mechanisms are the following:

- corrosion of the canister;
- leaching of the waste material;
- upward transport through the upper sediment layers to the lowest water layers;
- advection and diffusion through the water column;
- thermal effects on sediment or the water column;
- biological transport of incorporated isotopes across the seabed or upward through the water column.

In principle, the rates of all these processes are measurable or can be estimated. Regardless of the method chosen for emplacement of wastes in the seabed, calculation of breakthrough times (migration times) for each of these barriers must demonstrate that the waste will be contained for long periods of time.

The chapter will be organized in the following manner:

- Section 7.1, Ocean Characteristics

  This section describes the ocean environment and selects ocean regions which will be most suitable for waste repositories. A comparison of the relative merits of alternative ocean sites is made based on generic site selection criteria.

- Section 7.2, Emplacement Techniques

  This section discusses the possible methods of placing canisters at a proper depth in a sediment or rock layer.

- Section 7.3, Environmental and Health Considerations

  This section discusses the environmental and health aspects of seabed disposal.

- Section 7.3.1, Engineering and Environmental Barriers Against Waste Intrusion into Biosphere

  This section discusses migration mechanisms by which man may become exposed to radiation after its release from the deep sea emplacement site.

- Section 7.3.2, Research Needs

  This section identifies data required to understand the entire ocean-sediment waste system in order to adequately assess the feasibility of the seabed disposal concept.

- Section 7.3.3, Radiological Impact Assessment

  This section discusses the potential radiological impact to man and possible damage to the marine ecosystem from emplaced waste.

- Section 7.4, Economics

  This section provides data on costs for seabed disposal.

7.1 Ocean Characteristics

Several ocean provinces may contain possible locations for controlled emplacement of high-level radioactive waste under the sediments of the ocean floor. High-level wastes are the most difficult wastes to dispose of because of the combination of intense radiation and heat from the relatively short-lived isotopes and the great length of time required for the transuranic nuclides to decay. If one considers the ocean provinces on the basis of their overall suitability as disposal sites, it is possible to compare the relative characteristics of each province and apply the results of the comparison to select a potential disposal site. The criteria which have been used to evaluate disposal sites are as follows:[3,4]

- Temporal and Geological Stability: This may be estimated by observing the record of the past geological events held in the sediments;
- Inaccessibility: The areas selected should be as far removed as possible from the normal and expected activities of mankind;
- Lack of Resources: Waste disposal should not seriously interfere with the exploitation of resources;
- Permanence: Recovery of the waste material at a later date need not be a requirement;
- International Acceptability: If agreeable to all affected nations, seabed disposal may provide an international solution to nuclear waste disposal. As such, areas should be selected outside of direct national jurisdiction.

The ocean floors are divided into three principal physiographic provinces, each occupying about a third of the world's ocean area:

- Continental Margin, which includes continental shelf, inland seas, marginal plateaus, continental slope, and continental rises;
- Midoceanic Ridge, a global plate boundary which includes fracture zones, ridge flank and crest, and rift valley and mountains;
- Ocean Basin Floors, which include abyssal plains, abyssal hills, oceanic rises, and deep sea trenches (global plate boundary).

Characteristics of these three ocean provinces are summarized in Table 7-I.

A number of geological media have been considered for disposal beneath the seabed. Clay, shale, crystalline rocks of several kinds, and similar deep sea sediments are under consideration as prime disposal candidates or alternatives.

7.1.1 Continental Margin

The continental margins, located on the perimeter of the continents, represent the most dynamic environment of the ocean. Seasonal temperature changes in the water are high, chemical and biological processes are most variable, and the geology is most complex and unpredictable. Continental margins contain pools of hydrocarbons accessible with todays technology as well as most of the world's great fishing grounds. Surface sediments of these provinces change radically over short distances, ranging from hard rock to gravel to clay within only a few miles.

Continental margins may be characterized by:

- high resource value including food, mineral, hydrocarbons
- shallow water depth
- low geologic stability
- very strong and variable currents
- high sedimentation and erosion
- variable conditions (temporally and geographically)
- biological activity

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TABLE 7-1

CHARACTERISTICS OF THE OCEANIC PROVINCES

Physiographic Provinces of the Oceans	Mid-Oceanic Ridge				Ocean Basin Floor				Continental Margin			
	Fracture Zones	Flanks	Crest	Rift Valley	Seamounts Islands	Trenches	Abyssal Hills (Mid-Plate)	Abyssal Plains	Delta/Fans	Rise	Slope	Shelf
ENVIRONMENT												
Water depth (km)	4 to 6	3 to 5	0.5 to 2	1.5 to 3	0 to 3	7 to 11	5 to 6	4 to 6	0.1 to 5	3 to 5	0.2 to 3	0.1 to 0.5
Local relief (m)	1000's	100's	1000's	1000's	1000's	10 to 100	10 to 100	<5	10 to 100	5 to 10	100's	5 to 10
Regional slope (deg)	10's	2 to 5	10's	<5	10's	2 to 20	<1	<1/10	1/2 to 5	<1	4 to 10	<1
Bottom temp (C)	<3	2 to 4	3 to 5	5 to 15	2 to 20	1 to 3	<2	<2	3 to 5	1 to 3	3 to 5	2 to 20
Texture of bottom sediment	pebbles to sand	sand, silt, clay	boulders to sand	boulders to sand	gravel & sand	sand,silt,clay	clay	clay,gravel	sand,silt,clay	silt clay	silt clay	gravel & sand
Sediment thickness above igneous rock (km)	0 to 2	1/2 to 2	0 to 1/4	0	0 to 1/2	1 to 5	1/4 to 1/2	1/2 to 2	5 to 10	5 to 10	2 to 10	2 to 20
DYNAMIC PROCESSES												
Rate of sediment accumulation (cm 1000 yr)	0 to 10	2 to 4	0 to 5	0 to 1	<0 to 5	1 to 10	1	2 to 5	10 to 500	5 to 50	<0 to 3	<0 to 50
Currents: Non-tidal currents (cm sec)	10 to 20	3 to 5	10 to 50	10 to 20	5 to 25	2 to 10	2 to 10	<5	5 to 30	5 to 30	5 to 20	10 to 50
Currents: No. Near-bottom measurements	<10	<10	<10	<10	10's	<5	<10	10's	10's	100's	100's	1000's
Erosion or deposition	E	D	E	E	E	D	D	D	D	D	E	D E
Earthquake frequency	low to high	very low	moderate	very high	moderate	very high	very low	very low	very high to low, depending on position of plate boundary			
Biological activity	low	moderate	moderate	low	very high	moderate	very low	low	high	mod.	high	very high
Frequency of sediment failure	moderate	low	high	very low	moderate	high	very low	low	very high	mod.	very hi	low
Volcanic activity	moderate	low	high	very high	high	high	very low	very low	low	low	low	low
CHARACTERISTICS GERMANE TO SUB-SEA FLOOR DISPOSAL												
Geologic stability (predictability)	low	moderate	low	very low	very low	very low	high	moderate	very low	mod.	very low	very low
Area extent (km²x10⁶)	120				10	6	130	10 to 20	5 to 10	20	50	
Accessibility by man	low	low	high	moderate	very high	very low	very low	very low	moderate	low	mod.	very high
Present Value of Natural Resources, 10⁶ $ per Annum: Biologic	1 x 10³				1 x 10³				9 . 10			
Present Value of Natural Resources, 10⁶ $ per Annum: Sand,gravel	0				0				3 . 10			
Present Value of Natural Resources, 10⁶ $ per Annum: Mineral	0				0				6			
Present Value of Natural Resources, 10⁶ $ per Annum: Hydrocarbon	0				0				3 . 10⁴			

Reference: Seabed Disposal Program, A First-Year Report, December 1974.
Article entitled Seabed Disposal - Where to Look by W.P. Bishop, Sandia Laboratories, and C.D. Hollister, Wood's Hole Oceanographic Institution.

These dynamic elements are enough to rule out the margins because they do not meet the criteria of stability and isolation.

7.1.2 Mid-Oceanic Ridge

The Mid-Oceanic Ridge (MOR) forms the "construction" plate boundary of the ocean floor.[4] The center of the MOR is a hot, seismically active rift valley which continually extrudes new crust. Sediment thickness is typically too small to be detected. The center of the MOR may be characterized as follows:

- seismically and volcanically unstable, almost constant earthquakes
- without sedimentation
- topographically rough
- shallow in water depth
- having hot, molten basalt near surface

It is unlikely that the MOR center would be chosen as a suitable location for the disposal of large quantities of potentially hazardous waste.

The MOR is a global plate boundary and includes fracture zones, flanks, crests and rift valleys, and mountains. The flank areas are characterized by:

- high stability
- low resources potential
- inaccessibility

Based on these factors and the comparison provided in Table 7-I, the flanks of the MOR meet the criteria for acceptable waste disposal sites.

7.1.3 Ocean Basin Floor

The ocean basin floor is the deepest of the three provinces and includes the flat abyssal plains, abyssal hills, and deep-sea trenches. The flat abyssal plains have been created through deposition of sediments and debris from continental margins by strong currents. Sediments recovered from abyssal plains are typically silty clays mixed with coarsely graded layers of sand and gravel.

The abyssal hills were originally formed as extrusions of basalt from the MOR center. These regions are generally covered with 50 to 100 m of brown zeolitic clay overlying a few tens of meters of limestone. The concept of disposal under the ocean floor in the abyssal hills is attractive for several reasons:

- high geologic stability (seismically passive),
- invariant conditions (temporally and geographically),
- slow currents,
- low bio-productivity (low on surface, very low on bottom),
- limited resource potential

Deep-sea trenches and subduction zones are areas where, according to crustal global plate tectonics theory, one edge of a crustal lithospheric plate is moving under the other plate and down the earth's mantle into the asthenosphere (plastic zone) of the mantle. Sea trenches are among the less stable areas on the earth and undergo extensive changes in relatively short times. Deep sea trenches may be characterized as follows:

- seismically active
- volcanically active
- containing unstable sediments including slumping, sliding, and strong currents

These conditions do not meet the criterion of stability.

7.1.4 Criteria for Site Selection of Oceanic Provinces

Site selection criteria have been applied to the three ocean provinces to determine feasible locations for waste disposal. Following are the most important considerations:

- frequency of catastrophic events
- rates of natural processes
- predictability

Most of the data necessary to compare the three ocean provinces come from interpretations of past events by examining the properties of deep-sea sediment. The Deep Sea Drilling Project of the National Science Foundation and The Seabed Disposal Program of Sandia Laboratories and Woods Hole Oceanographic Institution have been instrumental in obtaining sediment data from numerous drilling experiments, seismic profiles (seismographs), and bottom sediment photographs. Interpretation of these data yields significant insight into the geologic stability and predictability over periods of million of years.

Comparing the major ocean provinces using the above criteria (see Table 7-I), it can be concluded that two ocean provinces are best suited for waste disposal. They are (in order of suitability) the abyssal hills and the flanks of the Mid-Oceanic Ridges.[3,4] Those areas which occur in the middle of the great oceanic gyres are especially attractive because of their low biological productivity. Thus, the areas in the middle of the tectonic plates and the middle of the gyres (mid-plate/mid-gyre) are best suited for waste disposal and have been the targets for further analysis. The mid-plate/mid-gyre region of the Pacific Ocean has been investigated as a potential site to perform further experiments and analysis of sediment samples. Core sample data have indicated that this region has a continuous record of millions of years of tranquility and geological stability. The Pacific Ocean mid-plate/ mid-gyre region is also characterized by unconsolidated clay sediments which make good sites for waste emplacement.

The Department of Energy (DOE) has supported the Seabed Emplacement Program to determine if any submarine geologic formation can contain radioactive wastes long enough for it to decay to innocuous level. More specific geological, geophysical, and oceanographic data are currently being obtained from site-specific studies at mid-plate/ mid-gyre areas, such as MPG-1[5] in the middle of the central North

Pacific, about 600 miles north of Hawaii. The Pacific mid-plate/mid-gyre region is a more suitable location for a disposal site than the mid-plate/mid-gyre region of the Atlantic for several reasons:[5]

- greater water depth in the Pacific Ocean;
- the Pacific has steady, deep, stable, and cold ocean currents capable of maintaining non-mixing conditions for perhaps a thousand years;
- the Pacific generally is believed to have geologically older sediments with a mineral composition (montmorillite and zeolite) containing higher distribution coefficients (Kd) than the Atlantic sediment (kaolinite and illite);
- the Pacific has greater distance from global plate boundaries and remoteness from man.

To extend the data base and further assess the mid-plate/mid-gyre environment, a second area (MPG-2) has been selected for sediment sampling and measuring. MPG-2 is located 700 miles northeast of MPG-1.[5]

Experiments are currently underway to establish the adequacy of the sediment to waste migration, especially with respect to the retention of radionuclides. Sediments that have adequate containment properties, such as brown oxidized clays, still have to be studied at sea to determine whether they can be found in sufficient thickness in MPG-type settings. Finally, it is necessary to determine in situ the physical and dynamic response of the sediments to emplacement and to establish sediment hole closure properties.[4,5]

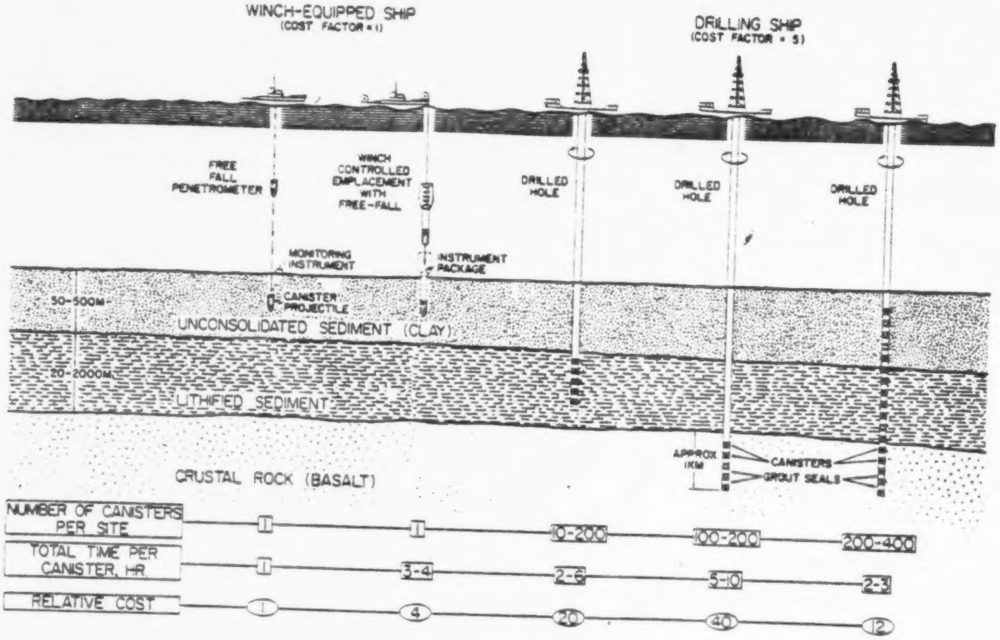
7.2 Emplacement Techniques[6,7,8]

Many possible methods of placing canisters at a specified depth in a deep sea sediment have been investigated (see Figure 7-1). These methods include controlled drilling from a surface ship and free-fall penetration (with a high velocity as driving force). Radioactive wastes could be emplaced either in the unconsolidated sediments such as oxydized red clays or in the underlying bedrock. The free-fall penetration technique would require a sediment with plastic properties which will collapse to fill the hole made by canister entry in a reasonable time.

An exact procedure for emplacement will not be chosen until it has been demonstrated that seabed disposal is feasible. However, it is necessary to consider one technique in order to assess the effects of emplacement on sediments. The technique chosen for analysis in this assessment is the free-fall penetration technique. The full spectrum of possible techniques should be studied, however, before a total emplacement system can be designated. A description of three emplacement methods is given below.

7.2.1 Free-Fall Penetration

In this emplacement method, the waste container would be dropped from a ship through the water column. A terminal velocity of 70 miles/hr would be reached before the canister would penetrate the clay sediments. Since the clay sediment is soft, it is expected


WINCH-EQUIPPED SHIP
(COST FACTOR = 1)
DRILLING SHIP
(COST FACTOR = 5)
FREE FALL PENETROMETER
WINCH CONTROLLED EMPLACEMENT WITH FREE-FALL
DRILLED HOLE
DRILLED HOLE
DRILLED HOLE
MONITORING INSTRUMENT
INSTRUMENT PACKAGE
50-500M
CANISTER PROJECTILE
UNCONSOLIDATED SEDIMENT (CLAY)
20-2000M
LITHIFIED SEDIMENT
APPROX 1KM
CANISTERS
GROUT SEALS
CRUSTAL ROCK (BASALT)
NUMBER OF CANISTERS PER SITE
1
1
10-200
100-200
200-400
TOTAL TIME PER CANISTER, HR
1
3-4
2-6
5-10
2-3
RELATIVE COST
1
4
20
40
12

SOURCE: Alternatives for Managing Wastes from Reactors and Post-Fission Operations in LWR Fuel Cycle, Volume 4, Waste Disposal, ERDA-76-43, May 1976.

FIGURE 7-1

ENGINEERING CONCEPTS FOR EMPLACEMENT OF RADIOACTIVE WASTE CANISTERS IN THE SEABED that penetration could exceed 30 meters. Monitoring instruments would be placed on the seabed floor to detect leaks. Canisters could be retrieved from the sediments, but this is not a goal of sediment emplacement. Laboratory studies indicate that closure of the emplacement cavity would occur immediately following canister penetration.

7.2.2 Winch-controlled Emplacement

In this option, the waste canister is attached to a drilling device designed to penetrate into the clay sediment. This device would either use momentum or some driving mechanism, such as vibration, to achieve penetration. One advantage of this method is that the canister could be immediately recovered in the event of a malfunction. However, laboratory studies indicate that there may be some hole closure problem associated with this method. If necessary, it may be possible to provide a sealant that could be left to fill the cavity above the canister when the drilling device is pulled out.

7.2.3 Drilled Holes

The technique for deep-sea drilling from a surface ship has been demonstrated by several marine research centers. This emplacement technique has the advantage that many canisters could be placed in a single bedded area at greater depths (100-500-meters) than other emplacement methods. As such, it will be necessary to develop a sealant which would fill the drilled cavity above and between the canisters.

To date, drilling techniques using sealant for seabed disposal have not been demonstrated.

7.3 Environmental and Health Considerations

7.3.1 Engineering and Environmental Barriers Against Waste Intrusion into the Biosphere

This section discusses the mechanisms by which radionuclides are transported from the emplacement through engineering and environmental barriers which retard migration to parts of the ocean of immediate significance to mankind. Because specific disposal sites have not been designated and because data on the rates of transport for all radionuclides are insufficient in some cases, the analysis of engineering and environmental barriers against waste intrusion into the biosphere contains many uncertainties. Until site specific data on transport mechanisms in deep-sea sediments and thermal effects on and by the canister are obtained and better understood, analysis of mechanisms by which radionuclides are transported will have to rely on generalized information on the ocean environment.

There are several mechanisms by which radionuclides are transported:[3,6,9,10]

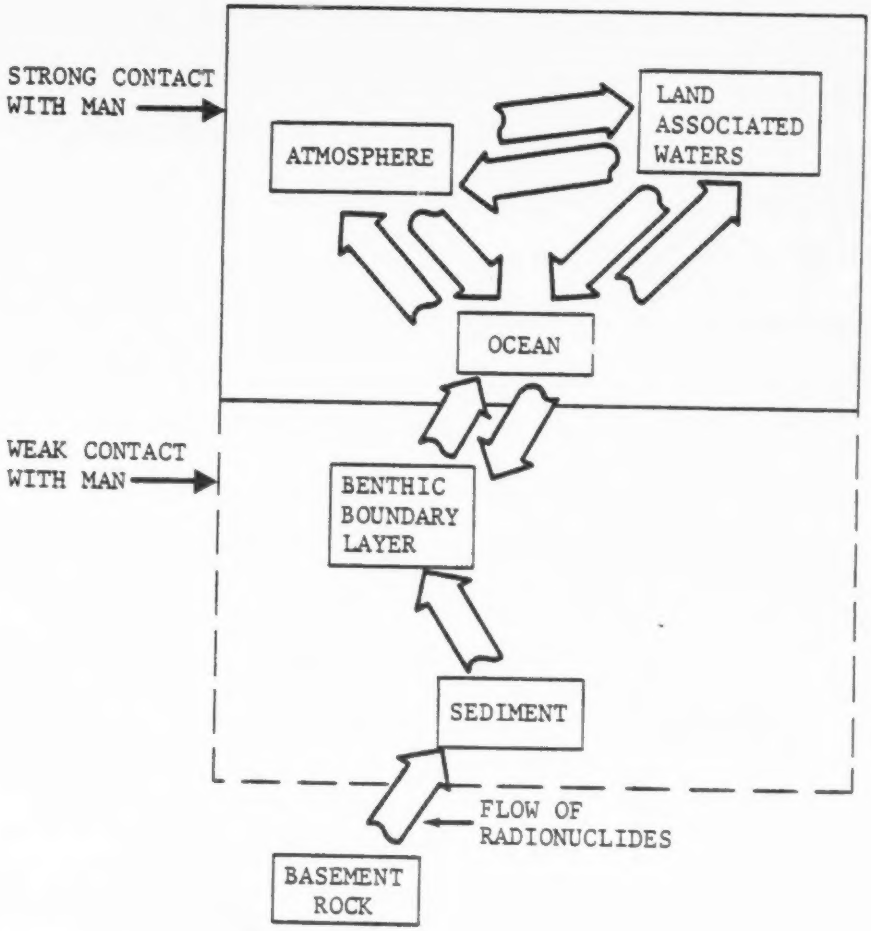
- corrosion of the canister
- leaching of the waste material
- upward transport through the upper sediment layers to the lowest water layers
- advection and diffusion through the water column
- thermally driven transport through the sediment or the water column

- biological transport of incorporated isotopes across the seabed or upward through the water column

In principle, the rates of all of these processes are measureable or capable of being estimated. Regardless of the method chosen for emplacement of wastes in the seabed, calculation of breakthrough times (migration times) for each of these barriers must demonstrate that the wastes will be contained for periods approaching geologic time scales. A diagram of the transport processes of radionuclides in the ocean which will be considered in this assessment is illustrated in Figure 7-2. This methodology forms the basis for discussion, the radiological impact to man, and ecological damage to the marine environment from seabed disposal.

7.3.1.1 Waste Form.[3,4,9] There are several considerations in providing engineering barriers against dispersion of radionuclides to the ocean environment. The first consideration is the specific waste form which is designed to prevent leaching of the waste material. The exact forms in which high-level radioactive and transuranic wastes will be packaged for seabed disposal are sensitive to the choice of fuel cycle, the physical characteristics, and the radiological properties of the waste material.

If the reprocessing option is implemented, the liquid waste produced during reprocessing of reactor fuel rods is basically a solution of radioactive and nonradioactive elements in nitric acid. The solution is very corrosive, generates large amounts of heat, and is highly radioactive. For waste disposal, these wastes have to


STRONG CONTACT WITH MAN
ATMOSPHERE
LAND ASSOCIATED WATERS
OCEAN
WEAK CONTACT WITH MAN
BENTHIC BOUNDARY LAYER
SEDIMENT
FLOW OF RADIONUCLIDES
BASEMENT ROCK

Reference: Alternatives for Managing Wastes From Reactors and Post-Fission Operations in the LWR Fuel Cycle, Volume 4: Alternatives for Waste Isolation and Disposal, ERDA-76-43.

FIGURE 7-2
TRANSPORT PROCESSES OF RADIONUCLIDES FROM SEABED DISPOSAL

be in suitable chemical forms which are stable even at the high temperatures caused by the heat from radioactive decay. The solubility of the chemical compounds in water must be as low as possible, so that even after final disposal, if there is any contact with water, the leach rate would be low. Present plans call for the solidification of the liquid waste by evaporation of the acid followed by incorporation in some stable material of high integrity such as concrete, glass, or zeolites. The percentage of radioactive waste that can be incorporated in the stable material depends on the chemical composition and nature of both materials. Not all of the fission products, particularly volatile radionuclides, can be incorporated into available types of material. For example, there is no technique currently available to fuse iodine compounds into glass. The problem of disposal of krypton-85 is difficult because krypton-85 (a noble gas) does not form a stable chemical compound. The only possible methods of disposal are storage at high pressure in cylinders and adsorption in some suitable porous material. In a companion study*, specific waste forms associated with the volatile radionuclides iodine-129, tritum, krypton-85, and carbon-14 are discussed in great detail.

The waste form itself forms the first barrier to migration. Several questions about the properties of these waste forms and their effectiveness in preventing dispersion are still unanswered. Exact leach rates for many of the fission products are not known because a final decision on the best types of waste forms has not been made. Few, if any, leach experiments have been carried out using solutions resembling sediment pore waters or at temperatures and pressure anticipated in the seabed after emplacement. If glass is used as the waste form, another question of concern is the long-term stability of the glass. The heat produced by the fission products during decay may convert the waste from a glass to a mass to tiny crystals. Devitrification may have the effect of speeding up the rate at which elements are released from the glass. Thus, the effectiveness of a glassy waste form may be very different if devitrification occurs in a few years rather than a few centuries.

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*Assessment of Waste Management of Volatile Radionuclides, The MITRE Corporation.

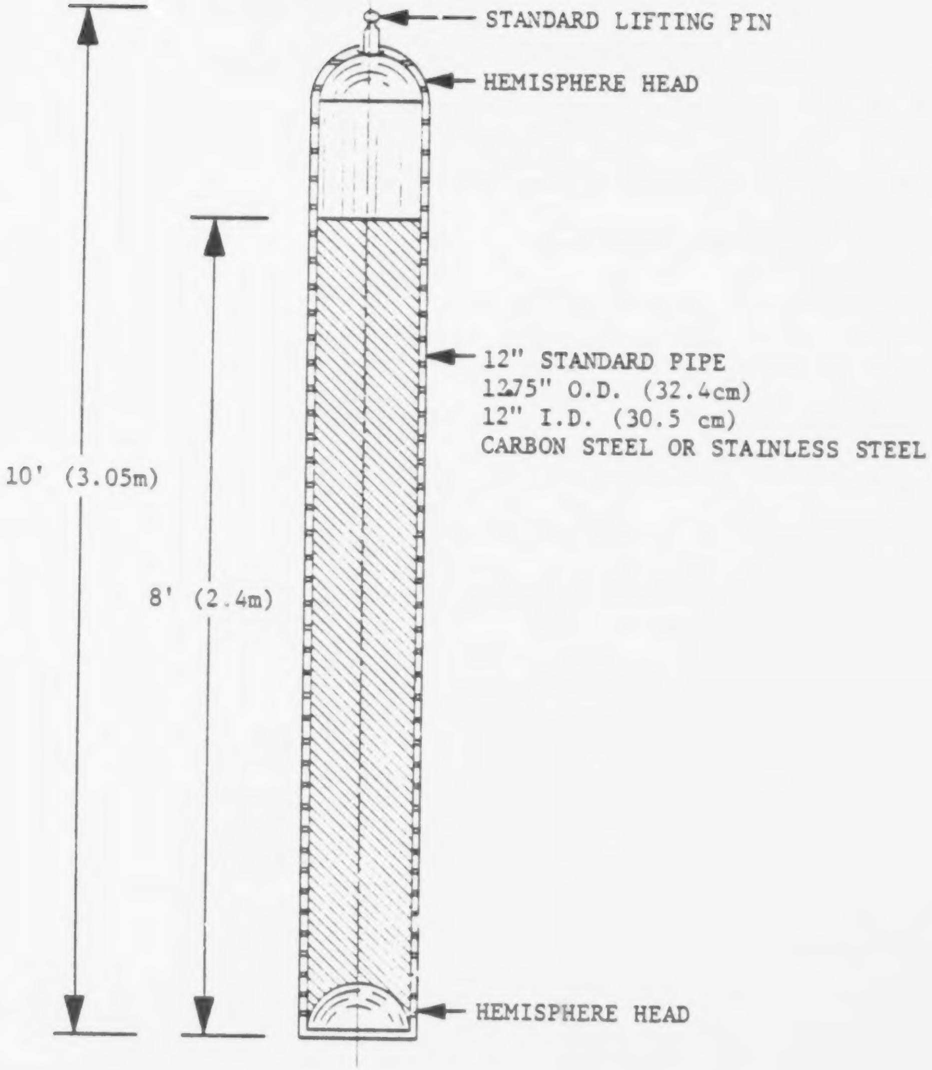
Because of the possibility of devitrification, a glass waste form may not confine radioactive elements for more than a thousand years. This duration is far less than the time period that is required for the longer-lived actinides to decay to innocuous levels. This period of time, however, may be long enough to allow the waste to dissipate most of its heat before the waste begins to interact with the surrounding sediments. Therefore, it is important to determine how effective the waste forms are in preventing isotope migration for the first several thousand years.

7.3.1.2 Canister.[3,4,9] High-level radioactive and transuranic contaminated waste, whether in solidified form from reprocessing or spent fuel form, will most likely be sealed in metal canisters. Glass or ceramic canisters are possible options, but may be less suitable for seabed disposal because of strength requirements for handling and shipment. Figure 7-3 shows a proposed standard canister for high-level, low-level, and intermediate-level wastes. The waste canister is the second barrier against dispersion of radionuclides to the ocean environment. The canisters will be designed to meet the following requirements:

- ability to dissipate the heat from newly packaged waste;
- long-term integrity of canister;
- ability to resist corrosion and leaching at high pressure and temperatures.

If seabed disposal is implemented, canisters will need to be designed to resist corrosion for a long time. Seawater (which is much like sediment pore water) is an extremely corrosive fluid. The only candidates for canister materials that appear suitable at present are titanium and zirconium alloys. Research to better understand the behavior of these materials in seawater and ocean sediments is being carried out at Sandia Laboratories. Several in situ corrosion experiments have also been conducted.

The corrosion of metals in marine environments limits their useful life and precludes the use of some materials which are attractive because of their low cost. The rates of corrosive attack have been documented for a large number of systems, even though the basic corrosion processes which occur are not well understood.


STANDARD LIFTING PIN
HEMISPHERE HEAD
12" STANDARD PIPE
12.75" O.D. (32.4cm)
12" I.D. (30.5 cm)
CARBON STEEL OR STAINLESS STEEL
10' (3.05m)
8' (2.4m)
HEMISPHERE HEAD

SOURCE: High-Level Nuclear Wastes in the Seabed?
Oceanus, Volume 20, Number 1, Winter 1977

FIGURE 7-3
THE PROPOSED STANDARD CANISTER

Experiments have shown that for many canister systems, the rates of localized corrosion (e.g., pitting, crevice corrosion) were high and may present a serious problem in the search for candidate materials which have extended lives (1000 years).

The best estimate at present is that materials such as zirconium and titanium alloys are capable of confining radioactive materials for a few thousand years. Again, this breakthrough time estimate is insufficient for the total containment that is required. Nevertheless as was the case for the waste forms, a thousand-year time period is long enough to allow the waste to dissipate most of its heat before it begins to interact with the surrounding sediment. This may have the net effect of reducing the possibility of rapid upward transport in convection currents that could be produced by heat dissipated from the canister.

To illustrate this point, a newly filled canister containing a mix of radionuclides, some with short-half lives and some with long-half lives, will give off 10 to 30 kilowatts of heat to the surrounding sediment.[9] A typical canister will radiate enough heat to raise the temperature to 600°C in the immediately surrounding sediment. This temperature will decrease to an undisturbed sediment temperature of about 0°C at a distance of 30 meters from the canister. At 600°C, strong thermal gradients are created which may cause rapid upward transport of leached radionuclides if these nuclides would breach containment immediately following emplacement. After about 1000 years, most of the short-lived radionuclides will have decayed and thus the total heat emitted from the canister will be reduced significantly. Based on the physical properties of deep-sea sediments, the temperature of the sediment immediately at the canister may be reduced from 600°C to about 200°C after 1000 years. This temperature reduction may have the net effect of reducing the possibility of rapid upward transport because of the reductions in the thermal gradients created at this temperature. This fact points out the importance of developing suitable waste forms and canisters which will be effective as barriers for approximately 1000 years.

The high temperatures in the vicinity of the waste canister could have the effect of fluidizing the entire sediment/pore water.[6] The canister could then sink through this viscous fluid to greater depths. Present knowledge is not adequate to predict whether this process would actually occur.

7.3.1.3 Sediment.[4,5,6,9,10,11,12]

Physical Properties

Information on the physical properties of deep sea sediments provides a very crucial part of the data necessary to evaluate transport mechanisms for radionuclides. A combination of spot sampling by drilling or coring and sub-bottom acoustic profiling techniques is used to obtain information on the physical properties of sediments.

Under the Seabed Disposal Program, several unconsolidated sediments have been sampled and examined to determine their physical properties and their appropriateness as a barrier to radionuclide migration. These sediments are as follows:

- oxidized red clay sediments
- calcium carbonate sediments
- silica sediments
- continental margin sediments

Oxidized red clay sediments have a number of physical properties that make them attractive as emplacement sites and have been the subject of studies on transport mechanisms in sediments. Oxidized red clays are extremely fine grain sediments with most particles less than 1 micron in diameter. As a result, they have low permeabilities (10^{-8} to 10^{-7} centimeters/second).[9] Oxidized red clays also have very large surface areas per unit volume of sediment. This is an important attribute in reactions between dissolved waste elements and clays, and in their ability to extract (sorb) metals from solutions.

To examine the barrier properties of oxidized red clays, data are being gathered in the following areas:

- distribution coefficients of sediments
- effects of heat on sediments (heat transfer properties)
- dynamic response of sediment to canister emplacement
- hole closure properties of clays
- biological and ecological implications of thermal waste heat on sediments

Distribution Coefficient and Retardation Factors[5,6,11,12]

Some fission products may react little or not at all with deep-sea sediments. These are expected to include tritium, krypton, technetium, iodine, and radon. The time it takes for these isotopes to migrate from the canister through the clay sediment to the sediment surface can be represented by the following:

$$T = d^2/C_d$$

where T = time [sec]
d = sediment depth [cm]
C_d = diffusion coefficient of element in sediment [cm^2/sec]

As an illustration, it would take iodine and tritium buried 100 meters below the deep-sea sediments approximately a million years to migrate to the ocean sediment interface. This is based on a diffusion coefficient of 3×10^{-6} square centimeters/second (which is an average value for deep sea sediments). For tritum, deep sea clays will certainly act as an effective barrier to migration because of its short half-life. Since iodine-129 has a half-life of 1.7×10^7 years, however, the clay sediments may not act as an adequate barrier to iodine migration. It will, however, reduce the cumulative time that the iodine exists in man's environment. Therefore, the sediment properties are a big factor for those radionuclides such as iodine-129 which have half-lives such that a significant quantity of the isotope still remains after canister and specific waste forms are no longer intact.

Most fission products, however, will enter into complex physio-chemical reactions with the deep-sea sediments by phenomena such as adsorption, ion exchange, and colloid filtration. These mechanisms are usually combined into one general term called sorption. Many waste elements with long half-lives such as plutonium, react with the clay sediment so that some of the element is sorbed to the sediment and some remains dissolved in the pore water. Sorption is expressed in terms of distribution coefficients, K_d, and is the ratio of the sorbed and dissolved concentration of isotope in the sediment. Because only the dissolved fraction diffuses through the sediment, the rate of diffusion of a reactive isotope is much slower than the rates for nonreactive elements such as iodine and tritium. The K_d values, are dependent on such parameters as pH of the water, the specific nuclide present, the concentration and type of dissolved ions, and temperature.

The effectiveness of deep sea sediments to act as a retarder for a particular condition is expressed as the retardation factor, R_d. For a particular radionuclide, R_d is defined as the ratio of the water velocity to the nuclide migration velocity (dimensionless term). The retardation factor is related to the distribution coefficients by the following relationships:

$$R_d = 1 + K_d \rho/E$$

where ρ = bulk density of the sediment
E = porosity (ratio of the volume occupied by pores to the total volume of the sediment).

The magnitude of radionuclide migration retardation that can be realized may be expressed by relating the velocity of ions moving through the sediment to the interstitial velocity of water flow by the following equation:

$$V_i = \frac{V_w}{R_d}$$

where V_i = velocity of the ionic isotopes

V_w = interstitial velocity of water flow

R_d = retardation factor

Estimated distribution coefficients (K_d), retardation factors (R_d), and relative transport rates of elements in soil to that in water (V_i/V_w) in a typical desert soil are shown in Table 7-II.

Although the distribution coefficients and retardation factors shown in Table 7-II are estimates of migration of radionuclides in typical desert soils, they do give a perspective and order of magnitude for K_d values for ocean sediments. Actual values for ocean sediments may differ substantially. The collection of data on the solutions formed by reactions between pore waters and specific radioactive wastes, and on the distribution coefficients for elements in such solutions, are task areas presently being undertaken at Sandia Laboratory. Distribution coefficients must be determined for all long-lived radionuclides as a function of sediment type. Some distribution coefficients are known for several candidate deep-sea sediments. These are summarized in Table 7-III.

From Tables 7-II and 7-III, the following is apparent:

- soil has greater retention for most of the long life radionuclides (actinides)than the short lived radionuclides,

TABLE 7-II

ESTIMATED DISTRIBUTION COEFFICIENTS (K_d) AND RETARDATION FACTORS (R_d) IN A TYPICAL DESERT SOIL

ELEMENT	K_d (ml/g)*	R_d†	Vi/Vw ‡
Tritium	0	1	1
Chlorine	0	1	1
Argon	0	1	1
Krypton	0	1	1
Technetium	0	1	1
Iodine	0	1	1
Astatine	0	1	1
Radon	0	1	1
Carbon	2	10	1×10^{-1}
Thallium	2	10	1×10^{-1}
Molybdenum	5	25	4×10^{-2}
Sodium	10	50	2×10^{-2}
Bismuth	10	50	2×10^{-2}
Calcium	15	100	1×10^{-2}
Antimony	15	100	1×10^{-2}
Neptunium	15	100	1×10^{-2}
Selenium	20	100	1×10^{-2}
Strontium	20	100	1×10^{-2}
Polonium	25	110	9×10^{-3}
Potassium	35	170	6×10^{-3}
Beryllium	75	330	3×10^{-3}
Cobalt	75	330	3×10^{-3}
Nickel	80	330	3×10^{-3}
Radium	100	500	2×10^{-3}
Rubidium	125	500	2×10^{-3}
Iron	150	3,300	3×10^{-4}
Cesium	200	1,000	1×10^{-3}
Francium	200	1,000	1×10^{-3}
Palladium	250	1,100	9×10^{-4}
Tin	250	1,100	9×10^{-4}
Promethium	600	2,500	4×10^{-4}
Samarium	600	2,500	4×10^{-4}
Europium	600	2,500	4×10^{-4}
Holmium	600	2,500	4×10^{-4}
Curium	600	3,300	3×10^{-4}
Berkelium	700	3,300	3×10^{-4}
Actinium	1,000	5,000	2×10^{-4}
Yttrium	2,000	10,000	1×10^{-4}
Zirconium	2,000	10,000	1×10^{-4}
Niobium	2,000	10,000	1×10^{-4}
Cadmium	2,000	10,000	1×10^{-4}
Plutonium	2,000	10,000	1×10^{-4}
Americium	2,000	10,000	1×10^{-4}
Lead	4,000	16,700	6×10^{-5}
Protactinium	4,000	16,700	6×10^{-5}
Thorium	15,000	50,000	2×10^{-5}

NOTES:

*Equilibrium distribution coefficients between water and soil.
†Retardation Factor (R_d) = Vw/Vi
‡Relative transport rate of elements in assumed soil to that in water.

Reference: Assessment of Geologic Site Selection Factors, Subtask C-1 Report, Arthur D. Little, Inc. November 1977.

TABLE 7-III

ESTIMATED DISTRIBUTION COEFFICIENTS (K_d) IN DEEP-SEA SEDIMENTS

ELEMENT	SEDIMENT	K_d
Sr	Montmorillite	104
	Kaolinite	15
	Illite	100
	Calcite	1
Cs	Montmorillite	4,400
	Kaolinite	45
	Illite	400
Pu	Montmorillite	630
	Kaolinite	352
	Illite	129
U	Illite	139

Source: L.L. Ames, D. Rai, "Radionuclides Interaction with Soil and Rock Media, "Vol. 1, EPA 520/6-78-007, 1978

- Considering Cs and Sr, clay minerals may have higher retention (higher K_d values) than the other earth materials. Comparing the common clay sediments, montmorillite has considerably higher K_d values than either kaolinite or illite. Montmorillite, however, contains weakly bonded interlayer water and the heat effects of the radioactive waste on this mineral would be an important consideration;

- The presence of biological activity and organic matter in the sediment is also an important factor in considering radionuclide retention by the sediment. Organisms in sediment would rework the sediment horizontally and vertically and redistribute radionuclides in the process. Organic compounds in solution offer little or no retention in the sediment and some are known to act as a transport mechanism for a radionuclide which would otherwise be retained by the sediment. Thus biological activity and the presence of organic compounds could offset the natural radionuclide retention capabilities of the sediments.

For an undisturbed system, deep-sea sediments may provide a satisfactory barrier to contain radioactive waste long enough for it to decay to innocuous levels. There may be some question as to the effectiveness of sediments as a barrier to longer-lived actinides. More research is necessary to assess the effectiveness of the sediment barrier after modification by heat and radiation of the canister. The hot canister may produce slow convection of the pore fluids leading to faster upward migration times of the radionuclides. The physical disruption of the canister emplacement is not expected to affect the sorption properties of the clays, but it may facilitate movement of the pore water.

Thermal Effects[6,10]

The response of sediments to heat generated by a waste container is crucial to an understanding of the effectiveness of deep-sea sediments as a barrier to radionuclide migration. Due to low thermal conductivity, high temperatures will exist around the canisters. After initial emplacement, the temperature of the surrounding sediment may be as high as 600°C. Substantial thermal gradients may exist around each container with temperatures declining to that of the surrounding sediments 10-20 meters away, (0°C). Such gradients give rise to upward pressure gradients which will cause water to migrate. This may well produce an upward flow of pore water away from the waste canister that will tend to carry the radionuclides toward the sea floor.

As previously noted, the high temperatures surrounding the waste canister could also have the effect of fluidizing the sediment/pore water. The canister, assuming a greater density than surrounding sediments, could sink to greater depths. In the event that failure of the canister released a sufficient quantity of heat-producing radioactive waste, such that the sediment/pore water was maintained in the fluidized state, it is conceivable that convective upward transport could occur. Present information on the physical behavior of the sediment is not available to determine if this process is possible.

Fortunately, the heat released by the radioactive waste will be reduced after several hundred years. For example, after a thousand years, the temperature of the surrounding sediment may be reduced to about 200°C because of radioactive decay of short-lived fission products. Thus, containment of the radioactive elements by the solid waste form and by the canister is very important to minimize any dispersion of these isotopes due to thermal effects for several hundred years after burial.

The effects of temperature on the distribution coefficient and retardation factors for the radioisotopes and their chemical compounds is also important in determining the isolation capababilitY of the deep sea sediments.

Rock Emplacement

Disposal in the deeper lithified sediments (at a depth of greater than 500 meters) is also being considered under the Seabed Disposal Program. Unlike the deep-sea sediments, the bedrock layers are susceptible to fracturing that could lead to fast migration of fluids along cracks. The fracturing is due primarily to higher shear strength and reduced plastic properties of these sediments. The transition down from soft deep-sea (clay) sediments to lithified deposits may be gradual or abrupt, and sometimes alternating layers of bedrock and soft clays are found. Data on lithified sediments below the sea floor have been obtained from Deep-Sea Drilling Project experiments.

Disposal within igneous rock beneath the ocean sediments has been considered, but only limited experimentation has been conducted. To date only a few holes have been drilled 500 meters or more into igneous rock by the Deep-Sea Drilling Project. From the few experiments conducted, the basement rock is comprised of the following:

> a layer of basaltic pillow lavas resulting from underwater eruption and rapid chilling of molten lava

- fractured blocks and breccia
- sediment-filled cavities and inter-layered sediments overlying quantities of basalt
- basaltic dykes at greater depths

The whole basement complex is cut by fractures and fissures at depths of 100 meters or more. Because the exact nature and predictability of these rocks is poorly known, neither basement rock nor the overlying lithified sediments are being considered as disposal sites at the present time.

7.3.1.4 Ocean.[3,4] The ocean water is likely to be a poor barrier for large quantities of released nuclides but provides some protection against inadvertent release of smaller amounts such as might be released from a single canister. Transport and dispersion through the ocean can occur due to a number of conditions:

- deep horizontal advection
- deep vertical mixing
- surface currents
- biological transport both horizontal and vertical
- thermal plume
- adsorption onto falling debris
- turbulent eddies

Material balance arguments and the age of the bottom water in the mid-plate/mid-gyre regions of the ocean indicate that the movement of dense water from the ocean bottom to areas where this water is returned to the surface layers takes from 1000 to 2000 years. Studies have indicated that the mixing time for the Pacific Ocean waters is 1000 to 1600 years or nearly twice as long as that of the Atlantic.

Knowledge of transport mechanisms of radionuclides through ocean water is far from complete. Data needs to be gathered in the following areas:

- bulk diffusion and advection coefficients
- effects of eddies and currents
- radionuclide scavaging by particulates in ocean columns
- biological transport through the food chain

Studies have indicated that the biological community either in the surface waters or on the bottom may provide a path for both horizontal and vertical transport.

7.3.1.5 Summary - Barrier Effectiveness for Waste Isolation. In Section 7.3.1, the emplacement of high-level wastes in geologic formations underlying the ocean floors was discussed relative to the technical feasibility of seabed disposal. The technical feasibility depends upon demonstrating that seabed disposal can contain radioactive waste long enough for it to decay to innocuous levels or not to exceed established radiation standards.

Physical and environmental barriers exist which may prevent migration of radionuclides to ocean areas of immediate significance to mankind. These mechanisms of breaching these barriers include the following:

- corrosion of the canister
- leaching of the waste material

- upward transport through the deep-sea sediments
- transport through the ocean columns

The rates of radionuclides migration for all of these processes have been estimated in Section 7.3.1. Because data on the rates of transport for all radionuclides in a varied sample of deep-sea sediments is insufficient in many cases, the estimates of mechanisms by which radionuclides are transported is based on generalized information of the ocean environment and will contain many uncertainties. The potential effectiveness of the barriers for waste isolation for several radionuclides is provided in Table 7-IV. For purposes of this estimate, it is assumed that canisters will provide an effective barrier for 1000 years, the waste form will exist for 1000 years, the sediment will delay radionuclide release to the ocean for 10^6 years, and the ocean will delay radionuclide entry to the human environment for 1000 years. Further research is obviously required to support these assumptions. Table 7-IV, therefore, only represents the potential barrier effectiveness.

7.3.2 Research Needs

The investment required to develop the necessary baseline information regarding ocean characteristics, emplacement techniques, and engineering and environmental barriers against waste intrusion into the biosphere from seabed disposal may be significant. There are large gaps in information required to understand the entire ocean-sediment waste system that is necessary to adequately assess the technical feasibility of seabed disposal.

TABLE 7-IV

POTENTIAL BARRIER EFFECTIVENESS FOR WASTE ISOLATION

Nuclide	$t_{1/2}$	$10t_{1/2}$*		Retardation Factor** R_d	Barriers Adequate to Allow Nuclide to Decay to Innocuous Levels***
Cs-134	2.05y	20.5y	y	1,000	A
Co-60	5.24y	52.4y	y		A
Kr-85	10.8y	1.08×10^2	y	1	A
H-3	12.3y	1.23×10^2	y	1	A
Pu-241	13.2y	1.32×10^2	y	10,000	A
Eu-154	16y	1.6×10^2	y	2,500	A
Sr-90	27.7y	2.77×10^2	y	100	A
Cs-137	30y	3.0×10^2	y	1,000	A
Cm-243	32y	3.2×10^2	y	10,000	A
Pu-238	86y	8.6×10^2	y	10,000	B
Sm-151	87y	8.7×10^2	y	2,500	B
Am-242M	1.5×10^2y	1.5×10^3	y	10,000	B
Am-241	4.58×10^2y	4.58×10^3	y	10,000	C
Ra-226	1.6×10^3y	1.6×10^4	y	500	C

TABLE 7-IV (Continued)

Nuclide	$t_{1/2}$	$10t_{1/2}$*	Retardation Factor** R_d	Barriers Adequate to Allow Nuclide to Decay to Innocuous Levels***
Cm-246	4.7×10^3y	4.7×10^4 y	3,300	C
C-14	5.7×10^3y	5.7×10^4 y	10	C
Pu-240	6.58×10^3y	6.58×10^4 y	10,000	C
Th-229	7.34×10^3y	7.34×10^4 y	50,000	C
Am-243	7.4×10^3y	7.4×10^4 y	10,000	C
Cm-245	9.3×10^3y	9.3×10^4 y	3,300	C
Pu-239	2.44×10^4y	2.44×10^5 y	10,000	C
Th-230	8×10^4y	8×10^5 y	50,000	C
U-233	1.62×10^5y	1.62×10^6 y	14,300	E
U-234	2.47×10^5y	2.47×10^6 y	14,300	E
Pu-242	3.79×10^5y	3.79×10^6 y	10,000	E
Cm-248	4.7×10^5y	4.7×10^6 y	3,300	E
Np-237	2.14×10^6y	2.14×10^7 y	100	E
Cm-247	1.6×10^7y	1.6×10^8 y	3,300	E
I-129	1.7×10^7y	1.7×10^8 y	1	E
U-236	2.39×10^7y	2.39×10^8 y	14,300	E

TABLE 7-IV (Concluded)

Nuclide	$t_{1/2}$	$10t_{1/2}$*	Retardation Factor** R_d	Barriers Adequate to Allow Nuclide to Decay to Innocuous Levels***
Pu-244	8×10^7y	8×10^8 y	10,000	E
U-235	7.1×10^8y	7.1×10^9 y	14,300	E
U-238	4.5×10^9y	4.5×10^{10} y	14,300	E
Th-232	1.4×10^{10}y	1.4×10^{11} y	50,000	E

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*99.9 percent decayed.

**Retardation Factors (R_d) represent estimates for each isotope in soils based on Analysis of Migration Potential, Subtask C-2 Report, Arthur D. Little, December 1977.

***A = canister
B = canister + waste form
C = canister + waste form + sediment
D = canister + waste form + sediment + ocean
E = the retardation factor will be significant in preventing the escape of the radionuclide

Innocuous levels mean less than 0.1 percent of the original activity remains.

7.3.2.1 Ecological Implications of Thermal Waste Heat. By affecting the physical/chemical conditions in its surroundings, the placement of radioactive wastes may induce ecological changes. Since waste disposal sites are areas of low biological productivity, the major effect of thermal waste heat is likely to be one of increased biological activity. Three major factors must be examined to assess the ecological implications of thermal waste heat:

- increase in biological activity may increase the rate at which the canisters are decomposed
- increase in biological activity may increase the rate at which radionuclides are transported through the sediments to the surface waters
- higher biological productivity which may result from increased temperatures may be counteracted by the biologically deleterious effects of ionizing radiation

7.3.2.2 Hole Closure.[3,4,6] Any emplacement procedures will disrupt the sediment layer of the ocean floor. In order to ensure safe emplacement, it is necessary to examine the response of clay sediments to canister emplacement, particularly the hole closure properties of clays. To prevent a decreased migration time of the clay barrier, it is essential that the hole created by emplacement of canisters be filled either with the same type of sediment or with a suitable sealant.

Laboratory and field experiments are underway at Sandia Laboratories to examine sediment behavior during and subsequent to penetration by waste canisters. These initial experiments indicate that closure of a completely penetrating projectile (such as the free fall emplacement method) would be immediate and total, while closure of a hole left open by an emplacement rod would be gradual.

7.3.2.3 Summary of Other Data Requirements. Areas which require further information to adequately assess the technical feasibility of seabed disposal, particularly its ability to act as a barrier to radionculide migration, include the following:

- information on the characteristics of ocean provinces to determine and establish their overall suitability as potential seabed disposal sites
- technological capabilities including transportation, shipment, and emplacement of wastes
- corrosion properties of canister materials at high temperatures and pressures
- leach rates for all radionuclides in proposed waste forms
- physical properties of deep-sea sediments
- sorption and distribution coefficients of deep-sea sediments
- retardation factors of sediments
- effects of thermal gradients on sediments (heat transfer properties)
- dynamic response of sediment to canister emplacement;
- transport processes of radionuclide in deep sea sediments including structural and chemical properties and driving forces
- transport processes in the water column, including diffusion currents, advection, biological (food web), and thermal plume

7.3.3 Radiological Impact Assessment[2,10,13,14]

This section will assess the potential radiological consequences to man of solidified high-level radioactive waste which is emplaced in deep-ocean sediments. The principal route of return to man that is considered in this assessment is via dispersion in the deep ocean, physical transport to the productive surface layers, incorporation in marine food chains, and consumption of contaminated seafoods by man. The consequent radiation exposure to man will be assessed in terms of both individual and collective doses. Radiation doses arising from concentration of beach sediments are also considered. In addition, operational and transportation accident risks will be discussed. The discussion presented here will rely heavily on the information provided in the previous sections.

It is intended that only broad conclusions be drawn from this section. In the course of discussion, those subject areas where more study or information is required to complete a radiological impact analysis will be highlighted. Most of the information contained in this section has been abstracted from two reports: Assessment of the Radiological Protection Aspects of Disposal of High Level Waste on the Ocean Floor, Grimwood and Webb, National Radiological Protection Board NRPB-R 48 (1976); and Consultants Meeting to Review the Radiological Basis of the Agency's Provisional Definition and Recommendations for the London Convention, International Atomic Energy Agency (IAEA), June 1977, London, England. These reports have attempted to assess radiation dose to man and possible damage to the marine ecosystem based on models which evaluate release rates and pathways of radionuclides to man. The reports are preliminary and contain large gaps in the information that would be necessary to complete an Environmental Impact Statement (EIS) on the radiological impacts of Seabed Disposal. No attempt has been made in either report to establish radiation protection standards although the criteria for such assessments have been addressed.

7.3.3.1 Source Term.[11] In the context of a rapidly expanding commercial nuclear program, concern is often expressed with regard to final disposal and potential for release of long-lived radionuclides to the environment. In the case of seabed disposal, Section 7.3.1 discussed the effectiveness of engineering and environmental barriers against waste intrusion into the oceans. A summary of the effectiveness of these barriers for waste isolation for several radionuclides was illustrated in Table 7-IV. The conclusion drawn from Table 7-IV is that the combination of environmental and engineering barriers may be inadequate to allow the radionuclides with long half-lives to decay to innocuous levels before spreading to productive surface layers of the ocean. Further, the assumptions upon which the barrier effectiveness is estimated are unproven. Therefore, earlier, though gradual, or later release of radionuclides may be expected.

When account is taken of the quantities of various nuclides emplaced in the seabed, their half-lives and their dispersibility, those radionuclides likely to be most significant in terms of radiation exposure to man and potential damage to the marine ecosystem are the volatile radionuclides (C-14, and I-129), and the long-lived actinides. To illustrate this point, the amount of each nuclide which would initially be present in a seabed repository is listed in Table 7-V for three cases, the throwaway fuel cycle, UO_2 recycle, and mixed oxide recycle. These amounts (expressed in grams) are based on 50,000 MTHM charged to the reactor, and a 10 year cool-off period. If all radioactive wastes from U.S. nuclear power production were buried in the sea, these initial quantities would be much larger, particularly if the current backlog of stored waste was buried in the

TABLE 7-V

RADIONUCLIDE AMOUNTS IN INITIAL SEABED REPOSITORY

Nuclide	Initial Mass in Place ** (g) Throwaway	 UO_2 Recycle	 Mixed Oxide Recycle
Cs-134	3.53×10^5	3.53×10^5	1.55×10^5
H-3	2.5×10^3	1.7×10^2	2.27×10^2
Pu-241	3.91×10^7	1.96×10^5	1.67×10^6
Eu-154	1.79×10^7	1.79×10^6	1.24×10^6
Sr-90	2.12×10^7	2.12×10^7	1.23×10^7
[illegible]37	4.96×10^7	4.96×10^7	5.10×10^7
Cm-243	3.6×10^3	3.6×10^3	3.62×10^4
Pu-238	6.50×10^6	2.14×10^5	3.87×10^6
Sm-151	2.14×10^6	2.14×10^6	3.62×10^6
Am-242m	2.26×10^1	2.26×10^1	4.95×10^2
Am-241	2.53×10^7	3.28×10^6	5.25×10^7
Ra-226	1.48×10^{-2}	5.50×10^{-3}	1.97×10^{-11}
Cm-246	1.48×10^3	1.48×10^3	3.18×10^5
C-14	1.71×10^2	1.71×10^2	2.17×10^1
Pu-240	1.11×10^8	9.10×10^5	1.68×10^7
Th-229	2.10×10^{-2}	9.60×10^{-3}	1.35×10^{-3}
Am-243	4.46×10^6	4.43×10^6	1.27×10^8
Cm-245	1.25×10^4	1.25×10^4	4.46×10^6

TABLE 7-V (Concluded)

RADIONUCLIDE AMOUNTS IN INITIAL SEABED REPOSITORY*

	Initial Mass in Place ** (g)		
Nuclide	Throwaway	UO_2 Recycle	Mixed Oxide Recycle
Pu-239	2.70×10^{8}	1.36×10^{6}	3.76×10^{6}
Th-230	2.91×10^{2}	5.85×10^{1}	2.19×10^{-7}
U-233	2.60×10^{2}	7.20×10^{1}	2.59×10^{2}
U-234	9.15×10^{6}	5.90×10^{4}	
Pu-242	2.24×10^{7}	1.12×10^{5}	2.57×10^{6}
Cm-248	1.41×10^{0}	1.41×10^{0}	2.44×10^{2}
Np-237	2.35×10^{7}	2.33×10^{7}	6.90×10^{6}
Cm-247	1.99×10^{1}	1.99×10^{1}	4.20×10^{3}
I-129	1.16×10^{7}	1.46×10^{4}	1.86×10^{4}
U-236	2.05×10^{8}	1.03×10^{6}	1.54×10^{5}
Pu-244	2.56×10^{-5}	2.38×10^{-5}	4.12×10^{-3}
U-235	4.02×10^{8}	2.01×10^{6}	9.50×10^{7}
U-238	4.72×10^{10}	2.36×10^{8}	2.26×10^{8}
Th-232	6.70×10^{1}	1.17×10^{1}	1.47×10^{0}

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*Information based on Analysis of Migration Potential, Subtask C-2 Report, Arthur D. Little, December 1977.

**Based on 50,000 MTHM charged to the reactors, and a 10-year cool off period.

sea. The Grimwood and Webb model, for example, assessed the potential radiological consequences of seabed disposal based on the total high-level waste which would be generated by a postulated world nuclear program of nuclear power production to the year 2000. The initial quantities listed in Table 7-V could easily be scaled to represent source terms which reflect the quantities of radioactive wastes from U.S. power production to the year 2000.

7.3.3.2 Environmental Pathways to Man.[2,3,4,10] After radioactive wastes migrate through the environmental and engineering barriers discussed in Section 7.3.2, the principal mechanisms by which radionuclides reach man are dispersion of waste material in the deep ocean, physical transport to productive surface layers; incorporation in marine food chains; and consumption of contaminated seafoods by man or exposure of man to contaminated beach sediments.

The lowest trophic level of the marine food chains is plankton. Phytoplankton constitutes the largest single source of biomass in the oceans and accumulates nutrient elements directly from the water. Light is necessary for photosynthesis by phytoplankton. If they are carried by currents to deeper waters, the lack of illumination will eventually cause their death. The major portion of the oceans in which incorporation of elements into the food chains occur is, therefore, the surface layers to a depth of 200 meters.

Zooplankton, the next higher trophic level, includes groups which are omnivores as well as carnivores. They derive most of their food either directly or indirectly from the phytoplankton layer. Zooplankton are found at all depths in the oceans but the extent of their vertical migrations is usually a few hundred meters and the biomass per unit volume is much lower at depths below a few hundred meters than in the surface layers.

The present marine food sources utilized directly by man come from higher nektonic trophic levels than the plankton. Both pelagic and benthic animals constitute important food sources, the most important both in terms of numbers and availability being the near-shore pelagic and benthic groups; the open-ocean pelagic groups being of intermediate value and the open-ocean benthic groups being by far the least important both at present and in future potential.

Marine organisms can accumulate radionuclides from food, water and suspended or deposited sediments. For phytoplankton, accumulation of activity occurs via direct uptake from the water in a similar manner to their uptake of nutrients. For zooplankton, the major source of radionuclides is the water but considering the relative quantities involved, it seems most of the uptake occurs via food, except for those nuclides which are only slightly concentrated in food. For other nekton, the majority of the activity is taken in via food rather than water.

The concentration of a radionuclide in a given organism may be greater or less than the concentration in the surrounding water, the ratio being known as the concentration factor. Although the uptake of radionuclides by organisms is a dynamic process which depends on many variables (including the physio-chemical state of the activity. temperature and salinity of the water, growth rate and physiological state of the organism), the concept of the concentration factor is meaningful in an environment such as ocean transport which changes slowly compared with the turnover rates of activity in the organisms comprising the food chain.

In order to calculate the eventual return of radioactivity to man via the marine food chains, it is necessary to estimate values for appropriate concentration factors and to define pathways and associated modes of exposure to man. For mixed marine plankton, a concentration factor of 10^4 is typical for many of the radionuclides, although concentration factors for many individual radionuclides are not available. Concentration factors for marine molluscs, crustacea, and fish are, in general, better known although, for some nuclides, there still is considerable uncertainty. Table 7-VI lists concentration factors for some of the major radionuclides. They have been taken from values given in several review documents and are thought to represent realistic values for edible flesh of these organisms. A list of pathways and modes of exposure to man for various radionuclides is shown in Table 7-VII.

7.3.3.3 Nuclides of Importance if Barriers Maintain Expected Integrity. As discussed in Section 7.3.1, engineering and environmental barriers exist which may prevent migration of radionuclides to

TABLE 7-VI

CONCENTRATION FACTORS

NUCLIDE	CONCENTRATION FACTOR FOR FISH	CONCENTRATION FACTOR FOR MOLLUSCS OR CRUSTACEA
H-3	1	1
Se	1000	300
Sr	1	3
Zr	30	100
Nb	30	100
Tc	10	100
Pd	10	300
Sn	1000	300
Sb	300	300
Te	10	1000
I	10	100
Cs	30	30
Pm	30	1000
Sm	30	1000
Eu	30	1000
Pb	300	100
Po	300	3000
Ra	100	1000
Ac	30	1000
Th	10000	1000
Pa	10	10
U	10	10
Np	10	10
Pu	10	300
Am	10	1000
Cm	10	1000

Source: Assessment of Radiological Protection Aspects of Disposal of High-Level Waste on the Ocean Floor, Grimwood and Webb, NRPB-R 48 (1976).

TABLE 7-VII

PATHWAYS TO MAN AND MODES OF EXPOSURES

NUCLIDE	PATHWAY	MODE OF EXPOSURE
H-3	Miscellaneous	All
C-14	Fish Consumption	Ingestion
	Crustacea Consumption	Ingestion
	Mollusk Consumption	Ingestion
	Seaweed Consumption	Ingestion
Co-60	Beach Dwellers	External Irradiation
Sr-90	Seaweed Consumption	Ingestion
Ru-106	Seaweed Consumption	Ingestion
I-129	Seaweed Consumption	Ingestion
I-131	Seaweed Consumption	Ingestion
Cs-134	Beach Dwellers	External Irradiation
Cs-135	Fish Consumption	Ingestion
Cs-137	Fish Consumption	Ingestion
Eu-154	Beach Dwellers	External Irradiation
Ra-226	Fish Consumption	Ingestion
Th-229	Fish Consumption	Ingestion
Th-230	Fish Consumption	Ingestion
Th-232	Fish Consumption	Ingestion
U-233	Seaweed Consumption	Ingestion
U-234	Seaweed Consumption	Ingestion
U-235	Seaweed Consumption	Ingestion
U-238	Seaweed Consumption	Ingestion
Np-237	Seaweed Consumption	Ingestion
Np-238	Fish Consumption	Ingestion
Pu-238	Seaweed Consumption	Ingestion
Pu-239	Seaweed Consumption	Ingestion
Pu-240	Seaweed Consumption	Ingestion
Pu-241	Seaweed Consumption	Ingestion
Pu-242	Seaweed Consumption	Ingestion
Am-241	Seaweed Consumption	Ingestion
Am-242	Seaweed Consumption	Ingestion
Am-243	Seaweed Consumption	Ingestion
Cm-242	Seaweed Consumption	Ingestion
Cm-243	Seaweed Consumption	Ingestion
Cm-244	Seaweed Consumption	Ingestion
Cm-245	Seaweed Consumption	Ingestion
Cm-248	Seaweed Consumption	Ingestion

Source: Consultants Meeting to Review the Radiological Basis of The Agency's Provisional Definition and Recommendations For The London Convention, IAEA June 13-17, 1977, IMCO Headquarters, London, England.

ocean areas long enough for them to decay to innocuous levels. These barriers included containment in metal canisters, solidified waste forms, deep-sea sediments, and the ocean column. The potential of the barriers' effectiveness for waste isolation for several fission products and actinides was illustrated in Table 7-IV.

If engineering and environmental barriers are assumed to be effective for 10^6 years, the elements identified by "E" on Table 7-IV are the waste isotopes which pose the greatest environmental impact. However, if these barriers are ineffective in preventing migration and dispersion in the deep ocean for 10^6 years, several other fission products with intermediate half-lives may escape containment and become dispersed into the deep ocean before they will have decayed to innocuous levels. This will have the net effect of increasing the radiological impacts to man.

Using the quantities of fission production and actinides initially present in a seabed repository (see Table 7-V), and assuming a barrier effectiveness of 10^6 years, the amount of actinides and fission products which could potentially be dispersed into the deep ocean is calculated in Table 7-VIII.

As shown in Table 7-VIII, several radionuclides will still be present in large quantities after 10^6 years of decay. These radionuclides include: U-234, U-235, U-236, U-238, Pu-242, Np-237, and I-129. If the initial quantities of wastes listed in Table 7-V were scaled to represent all radioactive wastes from U.S. nuclear power production,

TABLE 7-VIII

RADIONUCLIDE AMOUNTS AFTER 10^6 YEARS OF DECAY

Nuclide	t1/2	Mass in place after 10^6 years (g)*		
		Throwaway	UO_2 Recycle	Mixed Oxide Recycle
Cs-134	2.05y	0	0	0
H-3	12.3y	0	0	0
Pu-241	13.2y	0	0	0
Eu-154	16y	0	0	0
Sr-90	27.7y	0	0	0
Cs-137	30y	0	0	0
Cm-243	32y	0	0	0
Pu-238	86y	0	0	0
Sm-151	87y	0	0	0
Am-242m	1.5×10^2y	0	0	0
Am-241	4.58×10^2y	0	0	0
Ra-226	1.6×10^3y	0	0	0

TABLE 7-VIII (Continued)

RADIONUCLIDE AMOUNTS AFTER 10^6 YEARS OF DECAY

Nuclide	t1/2	Mass in place after 10^6 years (g)*		
		Throwaway	UO_2 Recycle	Mixed Oxide Recycle
Cm-246	4.7×10^3y	1.36×10^{-61}	1.36×10^{-61}	2.93×10^{-59}
C-14	5.7×10^3y	2.70×10^{-51}	2.70×10^{-51}	3.43×10^{-52}
Pu-240	6.58×10^3y	2.02×10^{-38}	1.66×10^{-40}	3.06×10^{-39}
Th-229	7.34×10^3y	2.08×10^{-43}	9.52×10^{-44}	1.33×10^{-44}
Am-243	7.4×10^3y	9.51×10^{-35}	9.51×10^{-35}	2.71×10^{-33}
Cm-245	9.3×10^3y	5.43×10^{-29}	5.43×10^{-29}	1.94×10^{-26}
Pu-239	2.44×10^4y	1.25×10^{-4}	6.29×10^{-7}	1.74×10^{-6}
Th-230	8×10^4y	5.03×10^{-2}	1.01×10^{-2}	3.79×10^{-11}
U-233	1.62×10^5y	4.99×10^0	9.99×10^{-1}	3.59×10^0
U-234	2.47×10^5y	5.53×10^5	3.57×10^3	—
Pu-242	3.79×10^5y	3.60×10^6	1.80×10^4	4.13×10^5
Cm-248	4.7×10^5y	3.23×10^{-1}	3.23×10^{-1}	5.59×10^1

TABLE 7-VIII (Concluded)

RADIONUCLIDE AMOUNTS AFTER 10^6 YEARS OF DECAY

Nuclide	t1/2	Mass in place after 10^6 years (g)*		
		Throwaway	UO_2 Recycle	Mixed Oxide Recycle
Np-237	2.14×10^6y	1.70×10^7	1.69×10^7	4.99×10^6
Cm-247	1.6×10^7y	1.91×10^1	1.91×10^1	4.02×10^3
I-129	1.7×10^7y	1.11×10^7	1.40×10^4	1.79×10^4
U-236	2.39×10^7y	1.99×10^8	1.00×10^4	1.50×10^5
Pu-244	8×10^7y	2.54×10^{-5}	2.36×10^{-5}	4.08×10^{-3}
U-235	7.1×10^8y	4.02×10^8	2.01×10^6	9.49×10^7
U-238	4.5×10^9y	4.72×10^{10}	2.36×10^8	2.26×10^8
Th-232	1.4×10^{10}y	6.70×10^1	1.17×10^1	1.47×10^0

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*Calculations of the mass in place after 10^6 years of decay is based on the decay formula $N = N_o e^{-\lambda t}$. The values for N_o were taken from the values from initial amounts of radionuclides in a seabed repository from Table 6.1.

particularly projections of accumulated waste through the year 2000, then the quantities of radioactive materials remaining after 10^6 years of decay would be significantly greater than the amounts shown in Table 7-VIII. If the radionuclides become widely dispersed in the deep ocean, then the radiological impacts on marine organisms may be less significant. However, if dispersion and physical transport of these wastes is localized, marine organisms as well as suspended sediments may receive large doses of radioactivity which, in turn, will be incorporated in marine food chains.

If engineering and environment barrier integrity is not maintained for 10^6 years, significant quantities of radionuclides with intermediate half-lives (i.e., 10^3-10^6 years) may be dispersed in the deep ocean and will undergo similar physical transport to productive layers of the ocean and, in turn, incorporated in marine food chains. The integrity of environmental barriers depends heavily on the transport mechanisms of radionuclides through the deep-sea sediments (retardation factors). As discussed in Section 7.3.1, research and experimentation on retardation factors for radionuclides in deep-sea sediments is being conducted but established data on these coefficients is currently not available.

7.3.3.4 Dose Assessment. This section discusses radiation exposure to man from seabed disposal in terms of both individual and collective doses. The data and results contained herein are abstracted from Assessment of the Radiological Protection Aspects of Disposal of High Level Waste on the Ocean Floor, Grimwood and Webb, NRPB-R 48 (1976). Two models were developed in this report which characterize the physical transport and mixing processes in the ocean, as well as incorporation in marine food chains and ultimate consumption of seafoods and radiation exposures to man. These models contain many assumptions and input data which will not be discussed here.

The following is a brief summary of the most significant findings of NRPB-R 48 and other conclusions from previous sections concerning the radiological implications to man from seabed disposal:

ICRP Recommendations

- In order to provide a basis for comparison with individual and collective dose estimates from seabed disposal, the maximum permissible annual intakes (MPAI) of activity by ingestion for individual members have been calculated for the principal radionuclides;
- ICRP recommended maximum permissible dose rates for external exposure are 0.5 rems y^{-1} for whole body irradiation and 3 rems y^{-1} for skin;
- ICRP have made no specific recommendations on collective dose limits.

Doses to Individuals via Critical Pathways

- The highest ratios of individual doses to the appropriate dose limit (or intake (I) to the MPAI) are for the potential routes involving consumption of deep-ocean fish or plankton. The maximum values of I/MPAI are of the order of 10-2 for both routes. The times at which these maximum values occur tend to be either short (50-100 years) or intermediate (500-2000 years). Critical organs are usually bone for Sr-90 and the actinides, and whole body for Cs-137.

- The highest predicted intakes by individuals in the critical group due to consumption of surface fish are of the order of 10^{-3} to 10^{-4} of the MPAI for fission products at 50 years, and may reach 10^{-5} of the MPAI for the actinides at 10^5 to 10^6 years.

- Similar types of results are obtained from the consumption of deep ocean fish except that the predicted intakes are one to two orders of magnitude higher than surface fish.

- For consumption of plankton, only Sr-90 has a significant predicted intake with a ratio of 4×10^{-2} at 100 years. Two actinides of comparable importance are Am-241 and Am-243.

- Postulated intakes from consumption of molluscs or crustacea are less than via the routes already mentioned.

- Intakes from drinking desalinated water are low.

- External doses from contamination of coastal sediments are comparable fractions of the dose limit for both skin & whole body irradiation. The highest doses in both categories are given by Cs-137 which would deliver 3×10^{-3} of the whole body dose limit and 7×10^{-4} of the skin dose limit. The calculated doses are at a maximum after only 100 years, and it is most unlikely in practice that the coastal sediments would become contaminated so quickly.

Collective Doses

- The only intake route actually established for collective doses is via consumption of surface fish. The nuclides that are responsible for the maximum individual doses give rise to the maximum collective doses and the same limitations on the accuracy of the available information also apply.

- The largest annual collective dose to the whole body due to consumption of surface fish is about 4×10^4 man rems at 10 years from Cs-137 and Sr-90 taken together. Collective doses to the whole body at longer times will be of the order of 10^2 to 10^3 man rems per year. Nuclides which contribute include Am-241, Am-243, Pb-210, Ra-225, Ra-226, and Sn-126.

- Collective doses to the critical organ, which is bone for most of the important radionuclides, are of the order of 10^5 man rems in the early stages due mainly to Sr-90, decreasing to 10^3 to 10^4 man rems at longer times from a number of different radionuclides.

- If plankton were to become established as a major direct food source comparable with fish, then the predicted whole body collective doses could be larger than those from consumption of surface fish. The maximum annual value of collective whole body dose is 2 x 10^6 man rems after 100 years due to Sr-90.

- The maximum annual whole body collective doses from consumption of desalinated water are small.

- External collective doses from contaminated sediments are of the order of 10^3 to 10^4 man rems for both skin and whole body in the early stages due to Sr-90 and Cs-137.

Comparison with Natural Levels of Activity & Levels Due to Fallout

- As an attempt to provide a further yardstick against which to compare the results of the calculations of water concentrations, and therefore the consequent doses, Table 7-IX lists the levels of natural and fallout activities for some of those nuclides known to be present in seawater. The levels of the same nuclides predicted by the modeling for the assumed input are also given. It can be seen that in no case does the prediction from the model exceed the natural level of the nuclide, and that in most cases the model predictions are orders of magnitude lower. Even for those short-lived nuclides such as Ra-225 which do not occur to a significant extent in nature, the model concentrations are less than the natural concentrations of any of the radionuclides listed. The highest concentration of any actinide predicted is comparable with the natural level of Ra-226. Most fission products do not occur in nature but are present in seawater as a result of fallout from nuclear weapons testing. The levels predicted by the models are comparable with these fallout levels.

- These comparisons are not intended as a justification of the introduction of high-level waste in the ocean, merely as an indication that although the numerical results predicted for individual or collective doses may appear high, they are considerably less than the current doses from natural activity in seawater would appear to be if calculated on the same basis.

TABLE 7-IX

LEVELS OF NATURAL AND FALLOUT RADIONUCLIDES IN SEA WATER

Nuclide	Natural activity in sea water (μCi cm^{-3})	Fallout activity in sea water (μCi cm^{-3})	Max. widespread surface water conc. predicted from postulated waste disposal operation (μCi cm^{-3}) (No containment)
Actinides:			
Pb-210	$(1 - 9) \times 10^{-11}$		2×10^{-15}
Po-210	1×10^{-10}		2×10^{-15}
Ra-226	1×10^{-10}		2×10^{-15}
Th-230	$(0.6 - 14) \times 10^{-13}$		2×10^{-17}
Th-234	1×10^{-9}		1×10^{-15}
U-234	1×10^{-9}		4×10^{-14}
U-238	1×10^{-9}		1×10^{-15}
Pu-239		1×10^{-12}	4×10^{-12}
Fission products:			
H-3	2×10^{-10}	5×10^{-8}	1×10^{-12}
Sr-90		3×10^{-10}	4×10^{-10}
I-129	3×10^{-11}		3×10^{-14}
Cs-137		5×10^{-10}	6×10^{-10}

Source: Grimwood and Webb, NRPB-R48, Reference 10

7.3.3.5 Operational & Transportation Risks. Seabed disposal involves the loading and shipment of high-level radioactive waste by sea to the emplacement site. Such shipments give rise to operational and transportation risks such as the loss of a canister into the sea. The potential radiological impacts arising from accidents during operation and transport of high-level waste to seabed disposal sites represent an integral part of the overall radiological impact assessment of the seabed disposal concept.

This section will briefly summarize possible operational and transportation accidents and risks from seabed disposal. The information presented has been abstracted from Evaluating The Loss Of A LWR Spent Fuel or Plutonium Shipping Package Into The Sea, Heaberlin & Baker, BNWL-SA-5744.

A more detailed description of the radiological impacts of transportation may be found in Final Environmental Statement On The Transportation of Radioactive Material By Air And Other Modes, NUREG-0170. Although this report addressed the environmental impacts resulting from the transport of radioactive material by air, many of the conclusions concerning transportation risks, particularly the assumptions and methodologies used, may be applicable to seabed disposal.

Pre-loss Conditions

- Two initial states for the shipping packages were considered prior to loss into the sea

(1) An undamaged package assumed to have its full design integrity
(2) Package damaged by a shipboard fire

The fire environment associated with commercial freighters is not well defined but data from Sandia indicates that fire temperatures in hydrocarbon fires (the type of fire most likely to occur) may reach averages of 1000°C. Other types of pre-loss damage, such as a collision by two vessels in a harbor, have not been considered.

- Since plutonium is not volatile and will not evolve as a gas even at high temperatures, no distinction has been made between a fire damaged and an undamaged package. An extended fire at 1000°C could, however, cause the canisters to rupture, but no significant release of plutonium is expected.

- In the case of spent fuel casks, after approximately 4 hours of high temperature fire, some fuel elements would begin to fail. This may lead to unanticipated releases at the ship fire.

Failure Mechanisms in the Sea

- Once the shipping package (damaged or undamaged) is lost into the sea, two failure mechanisms may take place:

  (1) hydrostatic pressure
  (2) corrosion

- Since it was assumed in Section 7.3.1 that canisters would be designed to withstand high pressures, only under the case of a damaged canister will there be any potential for canister collapse by hydrostatic pressure.

- Similarly, corrosion rates to canisters lost at sea will experience the same leakage rates as described in Section 4.2. However, the canister does not have the sediment barrier to protect against radionuclide migration. If the canister was damaged by fire prior to loss at sea, then the corrosion rates for canisters will increase.

Radiological Impact

- Radioactive materials released into the sea environment would disperse into a large volume of the ocean. Most of the radionuclides such as cesium and plutonium will be reconcentrated through the food chain to fish and invertebrates which could be eaten by man. The dose to a man from the consumption of fish, crustacea, and molluscs is highly dependent upon the concentration of radionuclides in the individual fish consumed.

- Table 7-X gives the population and average individual doses as the dose received over the period of intake and 50-year dose commitment for the plutonium package loss.
- Table 7-XI gives the doses for loss of a spent fuel cask.
- Only in the most severe case, that of a spent fuel cask in an extended fire, are the calculated radiation doses for the average exposed individual as high as natural background. All other cases had much smaller doses.

7.4 Economics

In Section 7.2, two basic emplacement techniques were described in detail:

- Free fall penetration
- Controlled drilling from a surface ship.

In the free-fall penetration method, high-level waste canisters of the types discussed in Section 7.3.1 would be dropped from a ship through the water column. A terminal velocity of 70 miles/hr would be reached at impact. This technique assumes that the medium for emplacement would be soft deep-sea (clay) sediments. It is projected from sample extraction experiments that these clays would be soft enough to allow a canister to penetrate from 30 m and more. Clearly, this method would be inappropriate if emplacement site surface layers are to be composed of underlying bedrock. If bedrock is the chosen medium, then the controlled drilling technique(s) from a surface ship would need to be employed.

TABLE 7-X

ESTIMATED DOSE AND DOSE COMMITMENT FROM MARINE FOOD CHAIN FOR LOSS OF PLUTONIUM PACKAGE AT SEA*

	DOSE DURING INTAKE	50-YEAR DOSE COMMITMENT
Population Dose (man-rem)	5.0	100
Average Individual Dose (rem)	5.7×10^{-6}	1.1×10^{-4}

*2.55 kg Pu per package - 1.5 wt% ^{238}Pu, 58 wt% ^{239}Pu, 24 wt% ^{240}Pu, 11 wt% ^{241}Pu, 4.9 wt% ^{242}Pu, 1.0 wt% ^{241}Am, typical recycle plutonium.

SOURCE: Evaluating The Loss of An LWR Spent Fuel or Plutonium Package into The Sea, Heaberlin & Baker, BNWL-St-5744.

TABLE 7-XI

ESTIMATED DOSE COMMITMENT FROM MARINE FOOD CHAIN FOR LOSS OF A SPENT FUEL SHIPPING CASK CONTAINING 3.1 MT OF URANIUM

LOCATION OF LOSS	INITIAL CONDITION	UNDAMAGED MINOR FIRE INTERMEDIATE FIRE	EXTENDED FIRE
Continental Shelf	Population (man-rem)	510	1×10^5
	Average Individual (rem)	5.9×10^{-4}	0.11
Deep Ocean	Population (man-rem)	<100	<100
	Average Individual (rem)	$<1.1 \times 10^{-4}$	$<1.1 \times 10^{-4}$

SOURCE: Evaluating The Loss of An LWR Spent Fuel or Plutonium Package into The Sea, Heaberlin & Baker, BNWL-St-5744.

In Section 7.3.1, the sediment chosen as the barrier against waste intrusion into the biosphere was soft deep-sea (clay) sediments. The reasons are three-fold:

(1) Several studies (previously mentioned in Section 7.3) have indicated that deep-sea (clay) sediments will act as effective barriers to radionuclide migration. Experimentation on distribution coefficients and retardation factors of radionuclides have been conducted for deep-sea sediments.

(2) Drilling techniques in several types of bedrock will create hole closure problems (see Section 7.3.2). Development of suitable sealants has not yet begun.

(3) The drilling techniques have not been demonstrated. Because the current policy is to dispose of high-level wastes in land-based repositories, funds have not been appropriated which would be adequate to test the accuracy and effectiveness of the drilling concepts.

Because of these facts, free-fall penetration is soft deep-sea sediments in the most likely form of emplacement to receive continued funding at this time. Therefore, the economics of seabed disposal will be presented using this concept as the base case (most likely case) for cost estimates. Cost estimates will also be provided for controlled drilling techniques, but these methods are less likely to be implemented.

7.4.1 Cost Estimates

Cost estimates for the free-fall penetration and for controlled drilling are given in Table 7-XII. As shown on Table 7-XII, the total costs for controlled drilling are more than twice as much as that shown for free-fall penetration.

TABLE 7-XII

SUMMARY OF COST DATA FOR SEABED DISPOSAL*

REFERENCE PLANT CAPITAL COSTS** ($ MILLION)	FREE-FALL PENETRATION	CONTROLLED DRILLING
1. Port of Embarkation	20	20
2. Sea Transport Vessel	100	100
3. Sea Drilling Platform	0	300
4. Platform for Free Fall	50	0
5. Drill Pipe and Casing	0	5
6. Monitoring Equipment	3	3
7. Shipping Cask (300)	45	45
TOTAL CAPITAL COSTS (rounded)	200	475
REFERENCE PLANT OPERATING COSTS* ($ MILLION/YR)**		
1. Port Operation	1	1
2. Sea Vessel Operation	8	8
3. Sea Platform Operation (either drilling or free-fall)	5	8
4. Drilling and Support Maintenance Operations	0	7
TOTAL OPERATING COSTS	14	24

*All costs are expressed in 1973 dollars.

**Capital costs are based on a 25-year plant lifetime, and a total capacity for storage of 45,625 MTHM.

***Plant operating costs are based on emplacing 1,825 MTHM/yr.

SOURCE: High Level Radioactive Waste Alternatives, Section 6: Seabed Disposal, BNWL-1900, Volume 3, May 1974.

Note: Cost estimates for free-fall penetration were changed slightly by MITRE staff to be consistent with other discussions in the report.

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1. David A. Deese, "Seabed Emplacement & Political Reality," Oceanus, Volume 20, Number 1, 1977.

2. "Consultants' Meeting to Review the Radiological Basis of the Agency's Provisional Definition and Recommendations for the London Convention," International Atomic Energy Agency (IAEA), June 1977, London, England.

3. "Release Pathways for Deep Seabed Disposal of Radioactive Wastes," Sandia Laboratories, IAEA-SM-198/34.

4. "Seabed Disposal Program - Annual Reports," Sandia Laboratories, SAND 74-0410, SAND 76-0256, SAND 77-1270, 1974, 1976, 1977, respectively.

5. Charles D. Hollister, "Seabed Disposal Option," Oceanus, Volume 20, Number 1, 1977.

6. Armand J. Silva, "Physical Process in Deep-Sea Clays," Oceanus, Volume 20, Number 1, 1977.

7. "High-Level Radioactive Waste Management Alternatives," Section 6, Seabed Disposal, BNWL-1900, Volume 3.

8. "Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle," Volume 4: Alternatives for Waste Isolation and Disposal, ERDA-76-43, 1976.

9. G. Ross Heath, "Barriers to Radionuclide Waste Migration," Oceanus, Volume 20, Number 1, Winter 1977.

10. P. D. Grimwood and G. A. M. Webb, "Assessment of the Radiological Protection Aspects of Disposal of High-Level Waste on the Ocean Floor," Natinal Radiological Protection Board, NRPB-R48, Oct. 1976.

11. "Technical Support for the Radiation Standards for High-Level Radioactive Waste Management," Subtask C-2, Draft, Arthur D. Little, Inc.

12. "Technical Support for the Radiation Standards for High-Level Radioactive Waste Management," Subtask C-1, Draft, Arthur D. Little. Inc.

13. "Technical Support for the Radiation Standards for High-Level Radioactive Waste Management, Subtask C-3, Draft, Arthur D. Little, Inc.

14. S. W. Heaberlin and D. A. Baker, "Evaluating the Loss of a LWR Spent Fuel or Plutonium Shipping Package into the Sea," BNWL-SA-5744, Battelle, 1976.

8.0 ICE SHEET DISPOSAL

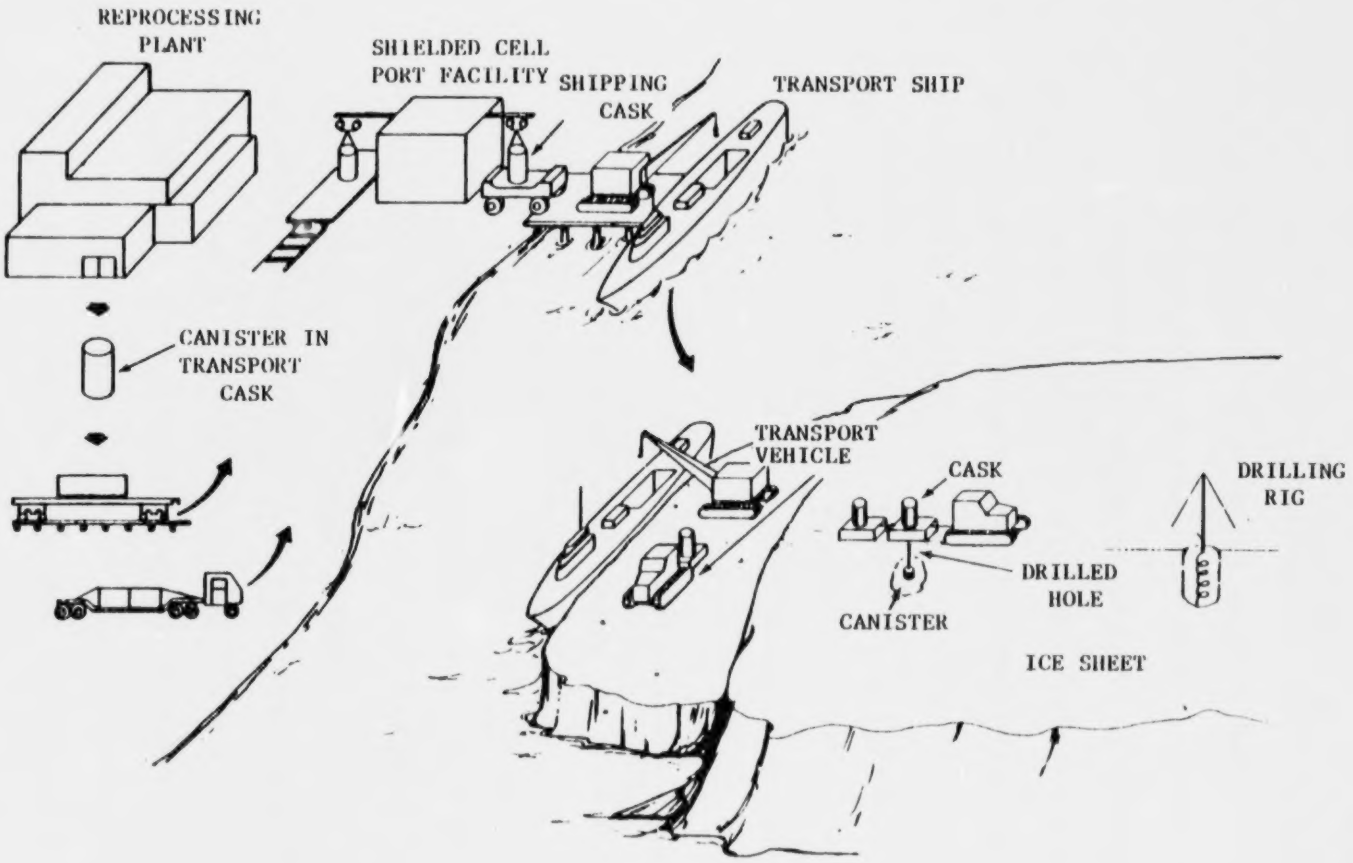
Continental ice sheets have been considered an alternative approach to the international solution for the final disposal of high-level radioactive waste. Theoretically, they could provide means for adequate geographic isolation of high-level radioactive waste from man's environment. However, the feasibility of ice sheets' long-term containment capailities is presently uncertain. These uncertainties exist in areas that have recently been reviewed by three international groups of glaciologists.[1,2,3] Their findings concluded that before ice sheets could be considered for waste disposal applications, certain areas of limited knowledge require further investigation:

- the evolutionary processes in ice sheets
- the relationships of ice sheets with climatic changes
- the nature of future climatic changes on the stability of the ice sheets

The following sections are a summary of the ice sheet disposal concept reported in references 4 and 5.

8.1 Descriptions of Ice Sheet Disposal Concepts

The ice sheet disposal alternative is considered in terms of the feasibility of three concepts discussed in the literature.[4,5] Waste disposal by any of the three concepts, if established, would be either in the Antarctica or Greenland ice sheets. A generalized schematic of the waste management operational requirements is shown in Figure 8-1. This schematic includes the basic system operations:


REPROCESSING PLANT
SHIELDED CELL PORT FACILITY
SHIPPING CASK
TRANSPORT SHIP
CANISTER IN TRANSPORT CASK
TRANSPORT VEHICLE
CASK
DRILLING RIG
DRILLED HOLE
CANISTER
ICE SHEET

Source: High-level Radioactive Waste Management Alternatives, BNWL-1900, Volume 3, Section 5, Ice Sheet Disposal, Richland, WA, May 1974.

FIGURE 8-1
SCHEMATIC OF OPERATIONS IN ICE SHEET DISPOSAL SYSTEMS FOR HIGH-LEVEL RADIOACTIVE WASTES

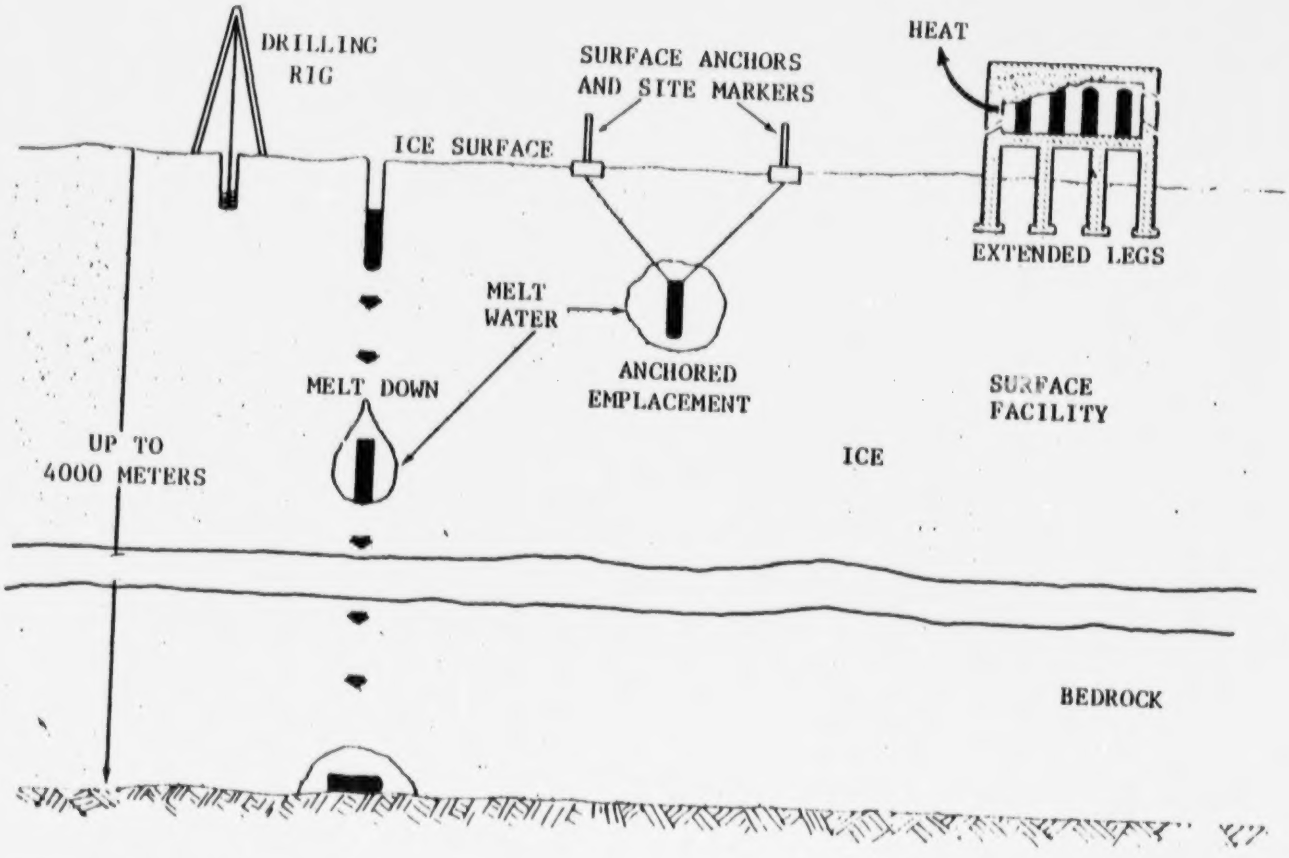
- transportation of solidified waste from the reprocessing plant or interim retrievable surface storage facility by truck, rail, or barge to embarkation ports;
- marine transport by specially designed ships during one-to-three-month periods of each year, with ice-breaker escorts near the ice sheets;
- a debarkation facility for unloading the waste canisters near the edge of the continent;
- the use of surface vehicles or aircraft for over-ice transport on a year-round basis;
- unloading and emplacing the waste canisters at the disposal site.

Ice sheet disposal of high-level radioactive waste would be done using one of the three concepts described in the following sections.

8.1.1 Meltdown or Free Flow Concept

The meltdown or free flow concept is shown in Figure 8-2.[2] In this concept, waste disposal is accomplished by selecting a suitable location in the ice sheets, predrilling a shallow hole, and eventually lowering the canister into the hole where it is allowed to melt down or free flow to the ice sheet basal.

Surface holes, predrilled to depths from 50 to 100 meters, serve as protective shielding from radiation during the initial operation phase of canister emplacement. To avoid individual canisters interfering with each other during descent and possible concentration at the ice sheet basal, it has been suggested that a spacing of about one kilometer apart will be required. Figure 8-2 shows a schematic of the meltdown or free flow concept.


DRILLING RIG
ICE SURFACE
SURFACE ANCHORS AND SITE MARKERS
HEAT
EXTENDED LEGS
MELT WATER
MELT DOWN
ANCHORED EMPLACEMENT
SURFACE FACILITY
UP TO 4000 METERS
ICE
BEDROCK

SOURCE: U.S. Energy Research and Development Administration, Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle, Volume 4 of 5, "Alternatives for Waste Isolation and Disposal," ERDA 76-43, Washgington, D.C., May 1976.

FIGURE 8-2
ICE SHEET DISPOSAL CONCEPTS

The canister meltdown rate is based on calculations from the penetration rates of thermal ice probes. It is estimated that the rate of descent for each canister would be on the order of 1.0 to 1.5 meters per day.[6] Assuming only vertical movement and an ice sheet 3000 meters thick, a period of 5 to 10 years is required for meltdown to the bedrock.

Another important factor in this concept is the design and shape of the canister. Adequate design and shape is important to assure a vertical path from surface to bedrock. In addition to the canister design and shape, the type of construction materials is important. Considerations for these materials should meet requirements for differences in ice sheet pressure and the possibility of saline water present at the basal.

There are also other options to this straightforward meltdown concept. Some appear more attractive from some viewpoints than others. For example, if the canister were so designed such that the contained waste and its the density was intermediate between those of water and ice, the rate of descent could be greatly decreased. The melt-down time would then approach that of the residence time of ice particles and by that time the canister would have become thermally inert.

8.1.2 Anchored Emplacement Concept

The anchored emplacement concept requires similar technology to the meltdown or free flow concept described above, the difference being that this concept allows for interim retrieval of the waste. Canister emplacement is accomplished also by drilling a hole in the ice sheet at a depth from 50 to 100 meters; cables 200 to 500 meters are attached to the canister before lowering it into the ice sheet. After meltdown, the canister is anchored at a depth of 200 to 500 meters by the anchor plates on or near the surface. The advantage of this concept, over the melt-down or free-flow concept, is that instrument leads attached to the lead cable could be used to monitor the condition of the canister during descent and emplacement. A period of 6 to 18 months is required for emplacement based on calculations from thermal ice probe rates.

Following emplacement, new snow and ice accumulating on the surface could eventually cover the anchor markers and present difficulties for their future recovery. The average height of snow and ice accumulating in the Antarctica and Greenland is about 5 to 10 cm/year and 20 cm/year, respectively. Future recovery of canisters for periods up to 200 to 400 years may be possible by using 20-meter high anchor markers. The approximate time for the entire system to reach bedrock at a typical site is estimated to be 30,000 years. During that time, the canisters and anchors would tend to follow the flow pattern of the ice.[4]

8.1.3 Surface Storage Facility Concept

This concept requires the use of large surface storage units constructed above the snow surface. The facilities will be supported by jack-up pilings or piers resting on load-bearing plates. Waste disposal would be accomplished by initially placing the waste canisters in cubicles inside the facility. Cooling of the canisters would be by air from natural draft. Elevation of the facility above the ice surface for as long as possible would provide for reduced snow drifting and heat dissipation. During this period the waste canisters would be retrievable. However, when the limit of the jack-up pilings is reached, the entire facility would act as a heat source and begin to meltdown through the ice sheet. It is estimated that such a facility could be maintained above the ice for a maximum of 400 years after construction.[4]

8.2 Status of Ice Sheet Technology Development

Current technology appears adequate for waste canister emplacement using the concepts previously described. Some uncertainties still exist in the technology and additional research is required. Further evaluation of transportation, logistics, and support facilities is needed to determine the feasibility of the technology. Improved means of inland transport of the waste over difficult and hazardous inland routes, and development of an efficient transportation system to carry the 20- to 25-ton casks require further evaluation. Areas of specific concern to a transportation system are fuel depots along the route and the means of fuel supplies.

8.2.1 Emplacement

Because the meltdown and anchored emplacement concepts are self-emplacing, little developmental research is needed for actual operation after the wastes arrive on site. Predrilled holes of 50 to 100 meters depth would be needed for initial emplacement. At this depth, the interconnecting air spaces in the ice have been sealed off into bubbles. Experimental holes up to 400 meters below the ice surface have been drilled using existing drilling equipment. These holes were "dry bored" and compressed air, served as the drilling fluid.

Because the surface storage concept would not require drilling, emplacement of the waste canisters would be accomplished by surface handling equipment on site. There is currently equipment available to handle casks without difficulty.

8.2.2 Transportation

Waste transportation from the embarkation ports to the areas of the disposal sites would be very difficult but not impossible. The ports would be designed for maximum safety, utilization, and acceptability. Consideration of docking facilities for large ships would be considered during dock design. The transport ships considered would be modifications of existing vessels. The ships would be equipped with the necessary safety features during construction. Current crude oil tankers are being built in the 400,000-dead weight ton class. Tankers of this capacity are larger than the ships required to transport the annual waste generated by a 5 MT/day reprocessing plant.

Although transportation appears adequate for transport from the embarkation port to the ice sheet margin, inland transport to the disposal site does present problems. These problems include slow travel, severe weather conditions, refueling, equipment maintenance, etc. Inland transportation would be necessary for the within-ice sheet concept to reach the most suitable location to gain access to areas of maximum thickness, stability, and as much isolation as possible. The distance inland that must be traveled (e.g., in the Antarctica) could be on the order of 1000 kilometers (600 miles).[4]

Unloading of casks at the continent margin would probably be done by crane or helicopter. Inland transport from this point could be accomplished by several methods. The reference study considered the use of surface sleds pulled by tracked vehicles, but this method has been abandoned by the U.S. in favor of aircraft as used to supply its permanent stations in the Antarctica. The average speed of the surface tracked vehicles is 3 to 6 kilometers per hour (2 to 4 mph), and considering trips of 1000 kilometers (600 miles) would require about 2 weeks travel per roundtrip.

Aircraft have been considered for inland transports, however, the use of aircraft is subject to limitations. Aircraft carrying payloads of up to 10 tons have been successfully used for transporting both personnel and supplies to Antarctica. Their use would involve high fuel consumption, probability of aircraft accidents, difficulty of navigation in severe weather conditions, and would require relatively drift-free landing areas at all times.

The final mode of inland transportation considered is Surface Effects Vehicles (SEV) such as hovercraft. SEV could be a possible means of transport, although they have not been tested in the high elevations of the Antarctica ice sheets (e.g., typical elevations are 460 meters at 16 kilometers, 1800 meters at 160 kilometers, and 2400 meters at 320 kilometers).[4]

The use of any type of surface vehicle to transport waste inland would require the establishment of a chain of fuel depots. Resupplying depots would probably be done using aircraft drop-offs. In this study, the conservation of fuel is considered a key item for any mode of shipment in the Antarctica.

8.3 Environmental Considerations

During several periods of the Pleistocene geologic epoch (approximately the last 2 to 3 million years), ice sheets covered about 30 percent of the earth's land mass. Only the ice sheets of the Antarctica and Greenland exist today which, together, cover about 11 percent of the earth's land mass. Together these two ice masses constitute the world's largest reservoir of fresh water (approximately 78 percent of the world's nonoceanic water).

8.3.1 Availability of Ice Sheet Data and Uncertainties

No information is presented in the literature that precludes the technical feasibility of high-level radioactive waste disposal in the continental ice sheets. The requirements for all waste management systems (i.e., transportation, logistics support, and emplacement) are available or could be made available through existing technology. However, the limitations of today's knowledge of the physics and history of ice sheets make the prediction of ice sheets stability uncertain for periods greater than a few thousand years. Verifications of theories that support ice sheet disposal would require many years of extensive new data collection and evaluation.

With regard to the limited data presently available, ice sheet disposal concept could offer potentially favorable features:

- geographic isolation
- relative isolation and containment of wastes by the ice in the event of leakage or canister failure
- low temperatures and high heat dissipation capacity
- relative safety from damage by storms, sabotage, and other hazards once the waste is emplaced

There are potentially unfavorable features for ice sheet disposal in general:

- extensive new data on all facets of ice sheet physics will have to be obtained
- the harsh environment and unpredictability of conditions on on ice sheets will present severe problems in establishing safe operations
- ice sheet areas are inaccessible during much of the year (8 to 11 months) because of storms, long periods of winter darkness, and freezing of surrounding seas
- monitoring and evaluating waste disposal operations would be difficult
- recovery from an unforeseen occurrence during transport to the disposal site would be difficult

8.3.2 Long-Term Containment

The capability of ice sheet to contain radioactive waste for long periods of time is presently speculative. Containment is highly dependent on the stability and physical properties of the ice sheet.

An analysis of the potential of canister failure upon emplacement in the ice sheets has been considered for the three disposal concepts. Providing that a canister failure should occur, the radioactive material contained would be in a potentially mobile system (i.e., the ice and water that may be present beneath it, either naturally or melted by the waste canisters). The probability of the waste eventually reaching man's environment, while in a hazardous form, depends greatly upon several factors of the system:

- o rates of motion within the ice sheet
- o the physical state and rates of ice flow
- o movement of meltwater at the base of the ice sheet
- o the long-term stability of the total ice sheet

8.3.2.1 Motions of Ice Sheets. Over the past few years, several measurements have been made to measure the surface motion rates of glacial ice.[6] Basically, these measurements have been done in the valley glaciers, ice shelves, and marginal areas of the ice sheets. The results of the measurements indicate a variation from centimeters per day to kilometers per year. Although mathematical models and theoretical studies have been made, the interior rates of ice sheet movement are essentially unknown.

8.3.2.2 Physical State and Rates of Ice Flow. Until recently the physical conditions at the base of the ice sheets were essentially unknown. Theoretically, some investigators suggest that in the central areas of the ice sheet which are sufficiently thick, melting could be occurring as the ice sheet moves as a rigid block sliding over underlying land, creating a bottom melting condition.[9-12]

Three general types of ice flow patterns are identified:

- o sheet flow--general outward movement of ice over a bed of low relief
- o stream flow--relatively rapid movement of valley glaciers and ice streams
- o ice shelf movement--general seaward movement of an ice shelf

The velocities of ice surface measured at a number of locations in various parts of the Antarctica are as follows:

- o sheet flow--0.05 to 0.15 meters (2 to 6 inches) per day
- o stream flow--0.3 to 2.6 meters (1 to 9 feet) per day
- o ice shelf movement--0.9 to 1.2 meters (3 to 4 feet) per day

In Greenland, "measurements of ice surface velocities are generally lower for sheet flow--as low as 0.1 centimeters (0.04 inch) per day, and as high as 27 meters (88 feet) per day for outflow glaciers."[4]

8.3.2.3 Meltwater at Base of Ice Sheet. Within the past few years, meltwater presence at the base of the ice sheet has been detected.[13] The dimensions of an ice sheet and its movement over the underlying material are controlled to some extent by the water layer. Measurements have been limited to a few bore holes which penetrated the bedrock. Here, meltwater detection has been done using remote-sensing techniques. It is known that water layers and under-ice lakes exist beneath parts of the Antarctica ice sheet. But the effect of its presence on ice motion theories and ice sheet stability has not been determined. However, various sources and methods have been proposed to account for the presence of a water layer at the ice sheet basal. These sources could be related several factors:

- o the geothermal heat flux that may raise the temperature to the melting point of the ice
- o frictional heat caused by the motion of the ice over the underlying rock may melt some of the ice
- o various combinations of geothermal heat flux and frictional heat may occur

The temperature of the water found at the ice-rock interface in a core hole drilled through the Antarctica at Byrd Station was estimated to be -1.6°C. Evidence found there indicates that the bottom surface of the ice was at the pressure melting point. Based on calculations, the water layer present was estimated to be at least 1 millimeter in thickness. A similar analysis was performed at Camp Century on the Greenland ice sheet. The water temperature found at the bottom of a hole drilled 1,375 meters was -13.0°C, which was well below the pressure melting point. Meltwater at the base of the ice sheet has been proposed as the cause of initiating the (East) Antarctica surges which were considered to initiate the northern hemisphere glaciations.

8.3.2.4 Long-Term Stability. The stability of the ice sheet for long-term containment is essential for waste disposal methods requiring waste isolation for periods of time of a few thousand years or longer. This, in turn, depends greatly on future snow accumulation rates compared to ice losses by melting, evaporation, formation of icebergs, and future world climatic changes. Present scientific opinion suggests that the Antarctica ice cap is growing or at least is stable. However, the future of its stability cannot be predicted from scientific interpretation of past climatic conditions from the available ice core. It is possible that the occurrence of manmade or natural climatic changes could affect the long-term stability of the ice sheets. The magnitude of such abrupt changes that might occur is presently unknown.

8.3.3 Characteristics of Waste Forms

The reference study considered only solidified waste forms such as borosilicate glass encapsulated in metal canisters. It may be stored in the interim for 5 to 10 years to allow some thermal decay, but will not need any further conditioning for disposal. At this age, each canister of waste will contain about 1 megacurie of radioactive material of a heat generating rate of about 3 kilowatts. This amount of heat generation is capable of raising the temperature of the waste to its melting point unless external cooling is provided. Adequate cooling of the casks would be necessary until the waste reaches the ice sheet disposal areas. At the disposal areas, the average ambient temperature is below 0°C and should provide adequate cooling. Upon emplacement of the canister in the ice sheet, an initial melt pool of about 70 meters in diameter will result. The hole will reseal because of the temperature of the surrounding ice and its plasticity.

8.3.4 Site Requirements

Site requirements will vary depending upon which disposal concept is selected. Requirements for the melt-down concept would require a location where the ice has the greatest thickness and stability. Such location would be as far from the coast as possible to assure maximum containment. Some investigators suggest that the best location for the melt-down concept would be near the top of buried ridges in the underlying bedrock where the ice thickness is thought to be one kilometer.[2] Here, the ice-rock interface temperature is considered lower than basin areas and lateral ice movement is minimal.

8.3.5 Radiological Risks

Only hypothetical dose calculations have been made for radionuclides released from an ice sheet disposal site into the ocean off the coast of Greenland. Based on assumptions that a failure occurs in the disposal system, the release of radionuclides into Greenland current of 8×10^6 m^3/sec would be 0.3 percent per year of the total inventory available and complete mixing would occur in the ocean rapidly. Human pathways are assumed to be mostly via fish consumption. The maximum dose was considered to be from an individual consuming 100 kg/yr of fish caught in these contaminated waters and is estimated to be 0.2 mrem/yr. (Also refers to Section 7.0 for discussion of radioactive releases to the ocean.)

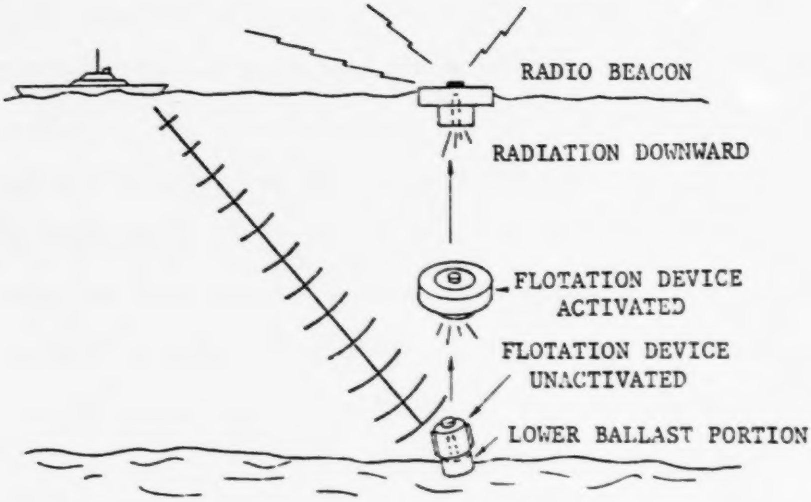
8.3.6 Accidental Risks and Consequences

The major accidental risks would be associated with transport at sea. In the event that a ship is sunk, the waste canisters could be equipped with flotation and other devices for recovery, as shown in Figure 8-3. This figure shows a typical recovery of a sunken cask at sea. When the cask sinks, a collar is activated which triggers the flotation device to raise the cask back to the sea surface. Relocation of the lost cask is done by radio signals given off after the cask reaches the surface.

The incident of a ship crashing into an ice pack and sinking could cause severe problems for canister recovery. During transport, the canisters would be enclosed in casks to prevent radiation and high temperature effects on the surrounding environment. Transport of waste is also discussed in Section 7.

8.3.7 Additional Data Requirements

Additional R&D requirements for ice sheet disposal are discussed from two perspectives: those related to obtaining basic information on ice sheets, and those related to the handling, transportation, and emplacement of the waste. Further studies are needed to adequately interpret the parts of the ice sheets, where the greatest thickness occurs. Ice motion measurements are also significant in predicting ice sheet long-term stability. Several measurements of surface motion have been made for parts of the surfaces of valley and outlet glaciers. Measurement of the interior motion is hindered by the lack of fixed landmarks. In order to obtain more accurate surface motion measurements, a minimum of 5 to 10 years of R&D would be necessary to provide meaningful data on the gross motion of ice sheets.


RADIO BEACON
RADIATION DOWNWARD
FLOTATION DEVICE ACTIVATED
FLOTATION DEVICE UNACTIVATED
LOWER BALLAST PORTION

Source: High-level Radioactive Waste Management Alternatives, BNWL-1900, Volume 3, Section 5, Ice Sheet Disposal, Richland, WA, May 1974.

FIGURE 8-3

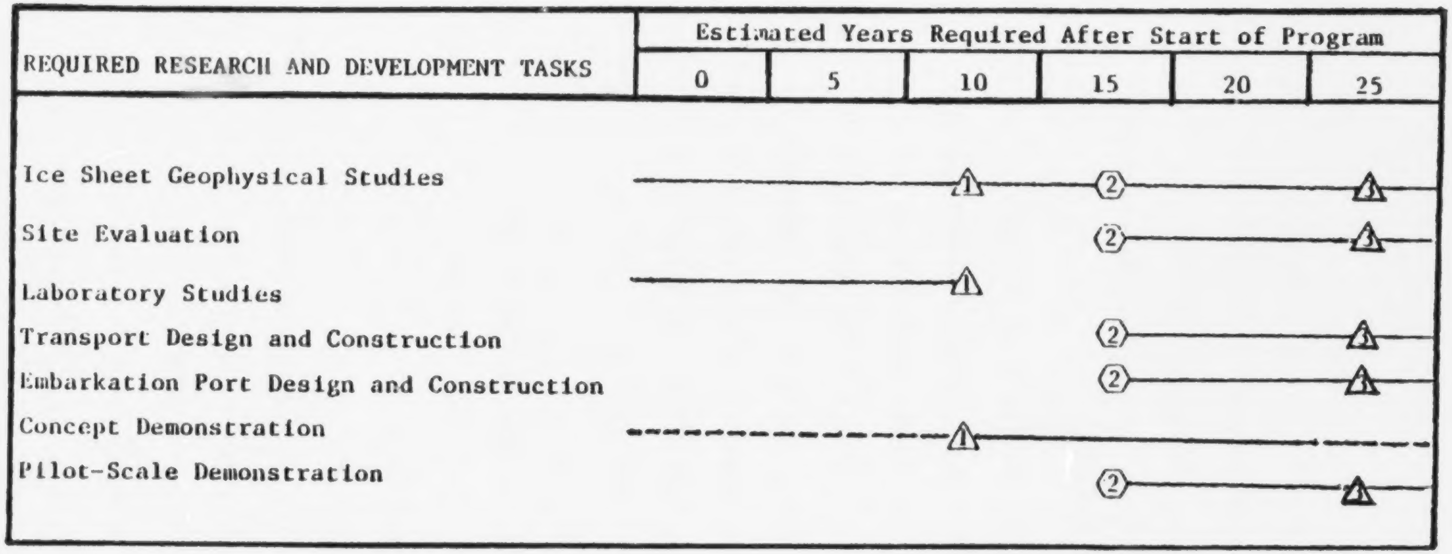
POTENTIAL CASK-CANISTER RECOVERY SYSTEM AT SEA

The stability of ice sheets (whether they will continuously exist in the future or whether they are expanding or shrinking) is presently unknown. To assure waste isolation for periods of hundreds of thousands of years, the present trend in the balance must be known to estimate future climatic conditions.

The estimated time required for expanded R&D programs to lead to the establishment of a commercial system for waste canister disposal in ice sheet is summarized in Figure 8-4. It is estimated that about 5 to 10 years would be required to select one of the three disposal concepts discussed after the program has been initiated. The minimum time required for the entire program is estimated to be 25 years to adequately evaluate ice sheets, in general, and conduct detailed studies necessary for specific site evaluations.

8.3.8 Summary

The ice sheet disposal concepts (assuming that operations are carried out as visualized) should have negligible environmental impact. The exception may be the potential impact on the ice sheet itself. Presently, it is difficult to assess the effects that waste canisters would have on ice sheets and of the interface conditions on the waste canisters until the physical conditions within the ice sheets and the ice-bedrock interface are better defined. In the meltdown and anchored-emplacement concepts, waste isolation from the environment can be assured as long as melting at the bottom of the ice sheets does not occur. The impacts on land, water, air, ecology, and aesthetics will be considered.



Key Milestones

1 Initial Data Developed - Tentative Site Selections

2 Decision on Disposal Concept - Final Site Selection

3 Start Routine Waste Disposal

Source: Modifications of High-level Radioactive Waste Management Alternatives, BNWL-1900, Volume 3, Section 5, Ice Sheet Disposal, Richland, WA, May 1974.

FIGURE 8-4

OVERALL RESEARCH AND DEVELOPMENT SCHEDULE - WASTE DISPOSAL IN ICE SHEET

Some land impacts would probably be experienced in connection with the embarkation port facility. An area of about one square kilometer would be required for the radioactive handling shielded cell and the loading dock facilities. The port facility would be equipped with its own separate water, power, and sewer systems to assure maximum safety.

The over-ice transport routes include an area at the edge of the ice sheet, ice shelf edge, and ice-free areas on land for unloading the shipping casks. Approximately six support and fueling stations will be required along the transport route to the disposal area. An additional 11,000 square kilometer area would be required for disposal of the output from a reference reprocessing plant of 5 MT/day.

Other possible land impacts considered in the reference study include accidental spills of fuel and the probability of fuel bladders rupturing during drop-offs. Rupture of the fuel bladders is considered to be a high risk because the fuel is capable of penetrating the snow and would reach the underlying ice where it will remain until evaporated or eventually becomes buried by additional snow.

Accidental spills could reach the ocean if the incident occurred near the edge of the ice sheet. Few, if any other impacts on water are expected, except for a marginal increase in temperature of the water used for once-through cooling of canisters during sea transport. The only other water uses would be for consumption by the 200 operating personnel, which would be obtained by melting the ice.

Air impacts would result from the combustion products of over-ice transport vehicles, support aircraft, and fuel consumed for heating the facilities at the disposal site. At present, the effects of these products are not considered a major problem. However, the accumulation of exhaust fumes and vapors over a long period of time may lead to temperature inversion and affect the weather pattern over the ice sheets. Altered weather patterns could conceivably influence the stability of the ice sheets.

Few, if any, ecological impacts are expected because the plant and animal life are confined mostly to the coastal areas. The construction of access routes and air traffic lanes could be done to avoid as much as possible the feeding, nesting, and mating areas of the birds and animals that inhabit the coastal areas.

Aesthetic impacts would be nil due to the remoteness of the area and lack of permanent residence population.

8.4 Capital and Operating Costs

The estimated capital and operating costs (1973 dollars) for the three ice sheet disposal concepts are summarized in Table 8-I.[4] Capital costs are primarily associated with transportation vehicles and equipment and are essentially the same for all three disposal concepts.

For the meltdown and anchored emplacement concepts, capital costs are estimated to be about $410 million to handle the waste from one reference fuel reprocessing plant. The associated operating costs are estimated at between $27 and $46 million per year. Capital costs for the surface storage concept are estimated to be about $415 million, with associated operating costs of about $23 million per year.

The total system unit charges are estimated to range between $19,800/MT for surface storage disposal and $23,500/MT for anchored emplacement (1973 dollars).[4] These charges include: reprocessing, 5-year interim liquid storage, solidification and containerization, 5-year interim solid storage, transport to the disposal site, and final emplacement.

8.5 Policy and Treaty Agreement

Because of treaty agreements, although the concept could be made feasible through further R&D, ice sheet disposal of radioactive waste is prohibited in Antarctica. However, Greenland (which is Danish territory) can be excluded from these restrictions.

TABLE 8-I

CAPITAL AND OPERATING COST ITEMS FOR ICE SHEET DISPOSAL

Capital costs, Million Dollars, for Meltdown and Anchored Emplacement:

1.	Construction of Embarkation Port Facility	20.0
2.	Sea Transport Vessel, Including Fully Equipped Hot Cell, 40-Ton Bridge Crane, etc.	100.0
3.	Two Ice Breakers @ #60 x 10^6	120.0
4.	Over-ice Transport Vehicles	53.0
5.	Drilling Rigs	12.0
6.	Monitoring Equipment	2.9
7.	Shipping Casks	45.0
8.	Aircraft	50.0
9.	Support Maintenance and In-Transit Facilities	5.0
	Total Capital Costs	407.9

Capital Costs, Million Dollars, for Surface Storage Facility:

1.	Construction of Embarkation Port Facility	20.0
2.	Sea Transport Vessel, Including Fully Equipped Hot Cell, 40-Ton Bridge Crane, etc.	100.0
3.	Two Ice Breakers @ #60 x 10^6	120.0
4.	Over-ice Transport Vehicles	53.0
5.	Surface Facility	20.0
6.	Monitoring Equipment	2.9
7.	Shipping Casks	45.0
8.	Aircraft	50.0
9.	Support Maintenance and In-Transit Facilities	5.0
	Total Capital Costs	415.9

Operating Costs, Per Year, Million Dollars:

Meltdown or Free Flow Concept

1.	Operation of Embarkation Facility	1.0
2.	Operation of Surface Facility with Hot Cell	8.1
3.	Transport Vehicles Operation	10.5
4.	Drilling Operations and In-Transit Facilities Operation	7.0
	Total Operating Cost Per Year	26.6

TABLE 8-I (Concluded)

CAPITAL AND OPERATING COST ITEMS FOR ICE SHEET DISPOSAL

Anchored Emplacement

1. Operation of Embarkation Facility	1.0
2. Operation of Surface Facility with Hot Cell	8.1
3. Transport Vehicle Operation	10.5
4. Surface Anchors, Cables, Chains	15.0
5. Drilling Operations and In-Transit Facilities Operation	7.0
Total Operating Costs Per Year	41.6

Surface Storage Facility

1. Operation of Embarkation Facility	1.0
2. Operation of Surface Facility with Hot Cell	8.1
3. Transport Vehicles Operation	10.5
4. Maintenance and In-Transit Facilities Operation	3.5
Total Operating Costs Per Year	23.1

(Cost in 1973 dollars)

SOURCE: High-level Radioactive Waste Management Alternatives, BNWL-1900, Volume 3, Section 5, Ice Sheet Disposal, Richland, WA, May 1974.

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4. High-Level Radioactive Waste Management Alternatives, BNWL-1900, Battelle Northwest, Richland, WA, Vols. 1 and 3, May 1974.

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9.0 CONTINENTAL GEOLOGIC WASTE DISPOSAL

Continental geologic disposal refers to those waste disposal methods related to interment of the waste in deep geologic formations on the continents. The deep-mined geological repository is, of course, included in this category, but is extensively discussed in other documents.[1] Although the deep-mined geologic repository is the most advanced and most studied concept, many alternative continental geologic disposal methods have been considered. While these alternative concepts may offer some advantages to deep-mined repositories in the form of engineering approach or economy, they also have a commonality of problems related to the assurance of the isolation of the waste from the environment. The containment problems, as discussed in Section 9.2, are sufficiently similiar that it might well be concluded that if the problems of deep-mined geologic repositories cannot be resolved, they are unlikely to be resolved for alternative geologic disposal methods. The exception to this might lie in the ultra-deep disposal methods where the greater depth of waste emplacement could provide an additional time barrier to transport into the environment. Technology development and cost will, however, be factors in the feasibility of such concepts. This section of the report presents the alternative disposal concepts in the following manner:

- Concept Description - A discussion of the engineering concepts
- Environmental Considerations - The geologic, hydrologic, and climatic considerations, and the pathways to the environment

- Technical Feasibility - A summary of the feasibility of the alternative concepts

9.1 Concept Description

The alternative concepts considered include the following:

- Solution-Mined Cavities
- Waste Disposal in a Matrix of Drilled Holes
- Waste Disposal in Super-Deep Holes
- Deep-Well Injection
- Hydrofracture
- Rock-Melting Concepts

Most of the alternative disposal concepts require the waste to be received in solid form. For a few of these concepts interim cooling may be required prior to final disposal. Figure 9-1 presents a basic flow diagram for solid waste disposal. If there is interim cooling, the steam and other off-gases to the condenser are passed through high efficiency filters in order to trap any radionuclides which may have escaped. A flow diagram for the liquid disposal process is shown in Figure 9-2. It must be noted that because of the serious problems that an accident in transporting high level liquid wastes would create, these concepts would most likely require that the reprocessing plant be at the repository site. The disposal of high-level and transuranic liquid waste is generally considered unacceptable due to the safety and containment problems involved. It may, however, be an acceptable method for low-level waste.

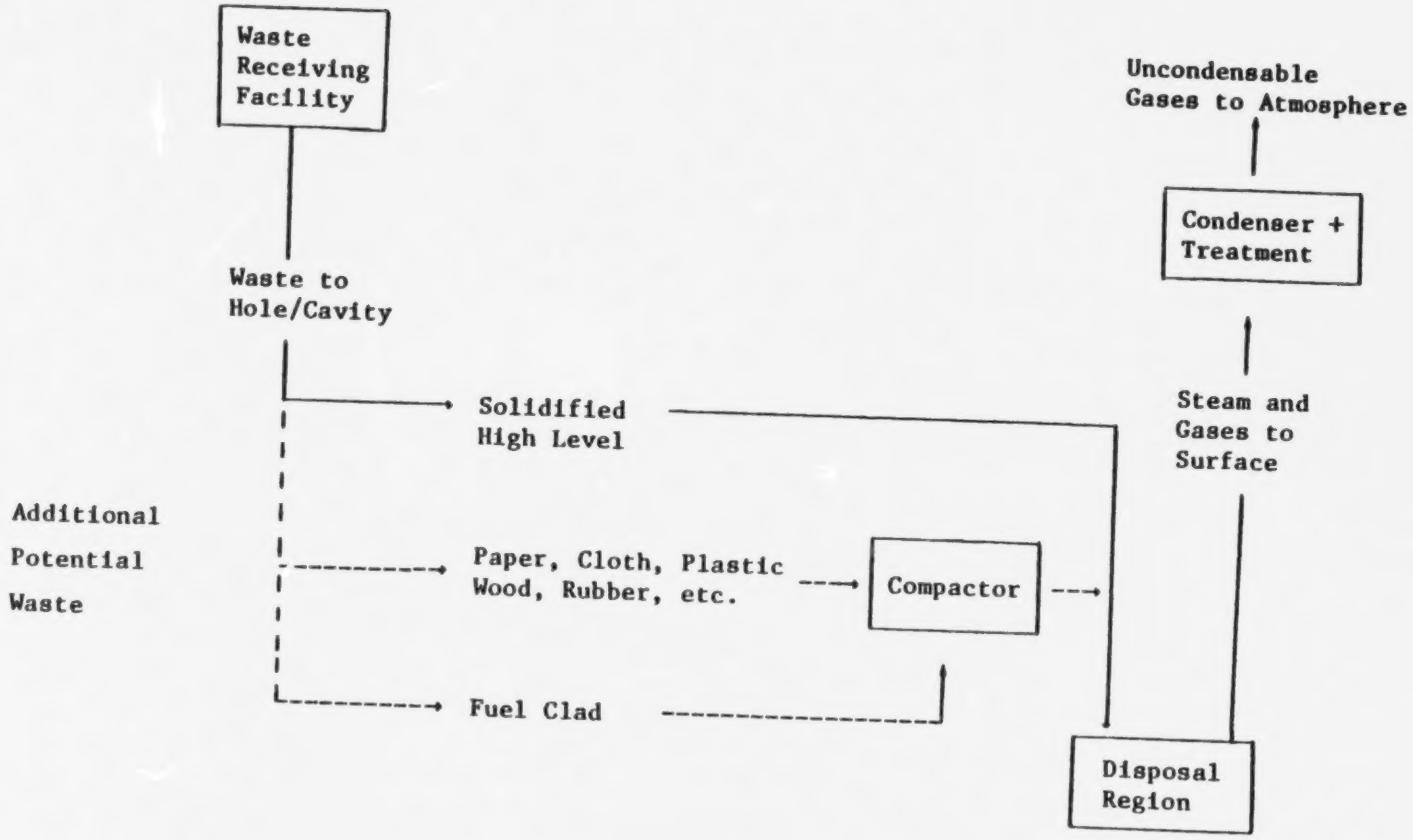

Waste Receiving Facility
Waste to Hole/Cavity
Additional Potential Waste
Solidified High Level
Paper, Cloth, Plastic Wood, Rubber, etc.
Fuel Clad
Compactor
Disposal Region
Steam and Gases to Surface
Condenser + Treatment
Uncondensable Gases to Atmosphere

FIGURE 9-1

FLOW DIAGRAM FOR EMPLACEMENT OF SOLIDIFIED WASTE

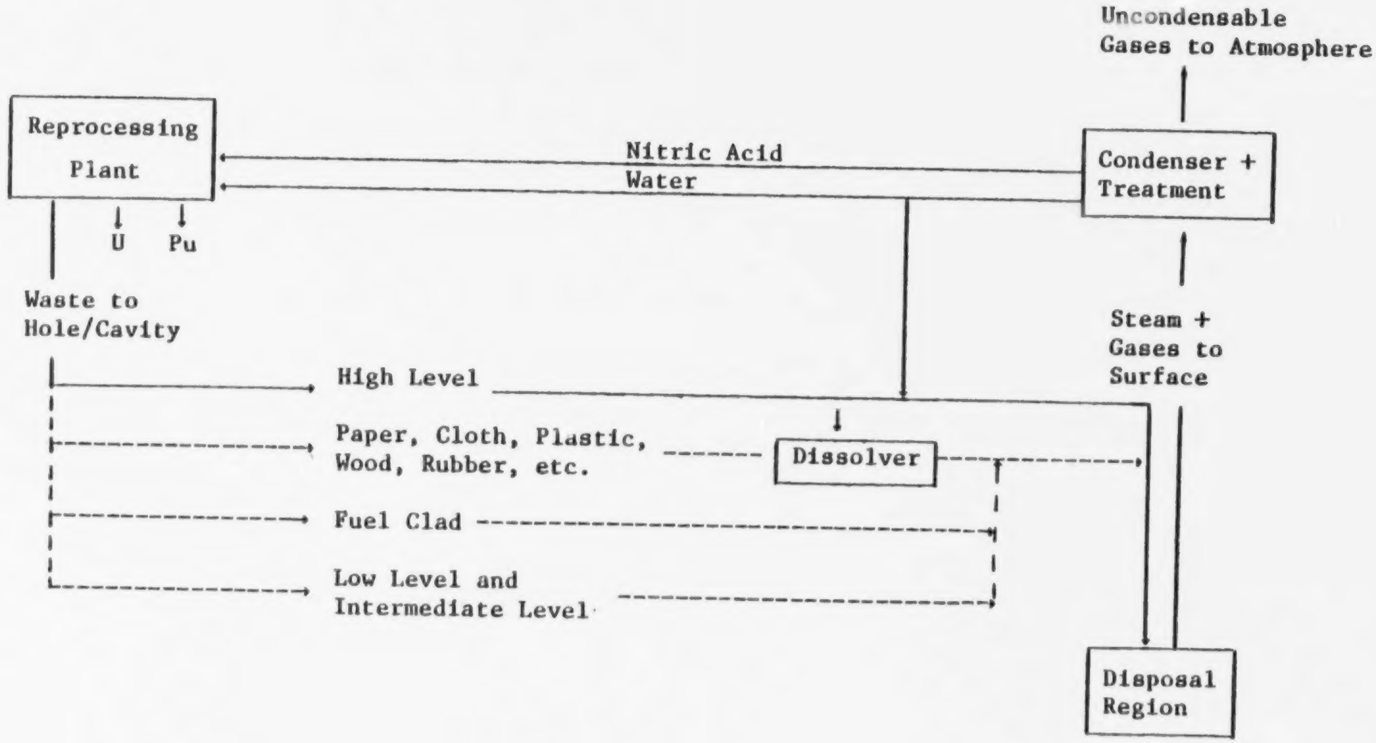

Uncondensable
Gases to Atmosphere
Reprocessing
Plant
Nitric Acid
Water
Condenser +
Treatment
U
Pu
Waste to
Hole/Cavity
Steam +
Gases to
Surface
High Level
Paper, Cloth, Plastic,
Wood, Rubber, etc.
Dissolver
Fuel Clad
Low Level and
Intermediate Level
Disposal
Region

FIGURE 9-2

FLOW DIAGRAM FOR EMPLACEMENT OF LIQUID WASTE

A brief description of each alternative concept follows. This will include method of emplacement, type of host rock which can be used, waste form, sealing from man's environment, depth of emplacement, and technical feasibility.

The Battelle, Pacific Northwest Laboratories report, ERDA 76-43, was a primary source of information contained in the following sections.

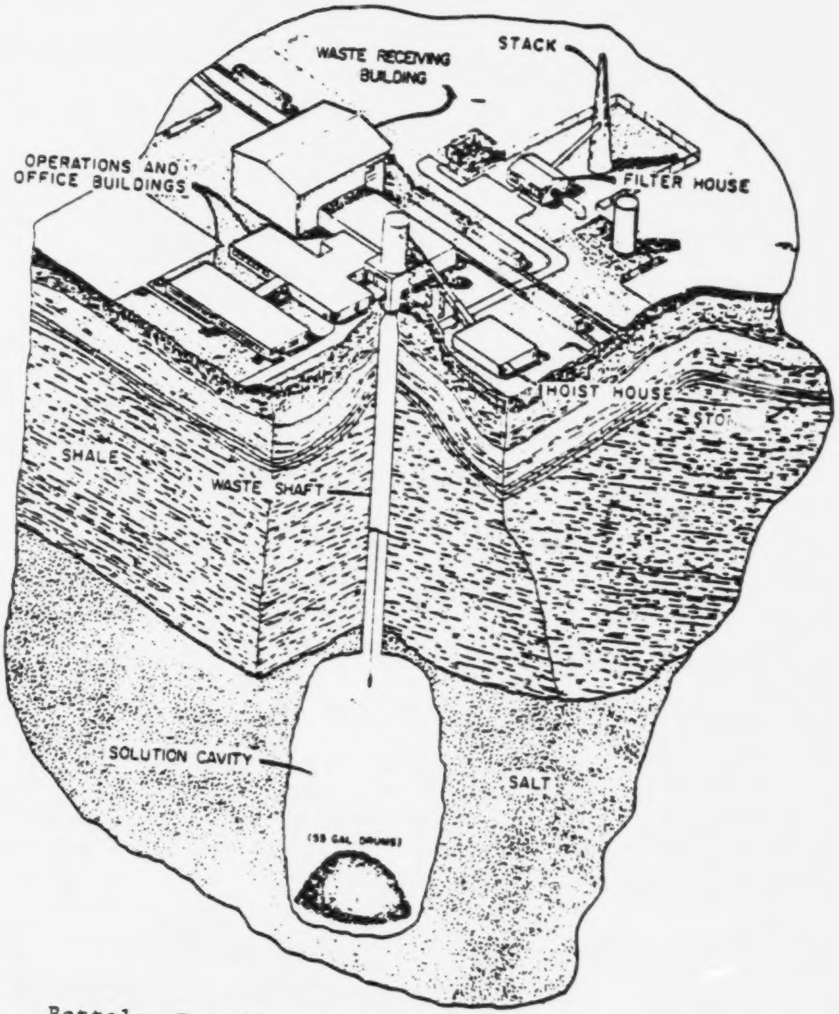
9.1.1 Solution-Mined Cavities

Salt is the only rock type in which solution-mining techniques can and are being used to construct large caverns. The current usage is mainly for storage of petroleum products. The technique consists of washing out the salt by fresh water action. The size and shape of the cavern can be controlled through manipulation of the fresh water flow, position of the inlet, location of the brine outflow pipe, the inert blanket, etc. The cavern can be constructed in salt which is in a dome, bedded, or anticlinal structure.[2] The technology for this concept is available now and would entail only surface facilities. However, such a disposal concept may have serious limitations.

The limitations may result from the type of emplacement itself. In this concept, the waste is received from the reprocessing plant in a solid form and, upon arrival, is unloaded from the shipment casks by remotely operated equipment. The waste is then moved into hot cells for inspection, monitoring, decontamination, repair (patching overpacking), and, finally, still using remote equipment, loaded into the hoisting facility. After the canister is placed in the hoisting device, it is lowered into the cavern (300 - 3,000 meters below surface) until it is near the bottom; it is then allowed to fall onto a random pile of canisters. Figure 9-3 shows a generalized concept of a solution-mined storage facility. The random placement of canisters presents a problem if there are high-heat generating materials within the canisters. The salt host rock may not be able to dissipate the heat away quickly enough to prevent melting and consequent flow of salt. This concept is therefore limited to handling only low-heat generating transuranic wastes. There are additional problems. Little is known about the stability of the caverns once they are dried out. There are also questions on the optimal size and shape of caverns to assure the greatest stability as well as the best drying method to be used. There is also the question of retrievability. "Fishing" by grapple for canisters is not a demonstrated retrieval method and disposal of high-gamma transuranics in the cavern and uncertain cavern stability would preclude direct loading of canisters onto the hoist by men lowered into the cavern.

9.1.2 Waste Disposal in a Matrix of Drilled Holes

In this concept, a matrix of holes about 1 meter in diameter would be drilled into a thick, tight geologic formation with no cracks, fractures, faults, etc., to permit water to circulate. These holes would be drilled to a depth of 300 to 6000 meters. Salt domes, bedded salt, argillaceous, intrusive igneous, and metamorphic


WASTE RECEIVING BUILDING
STACK
OPERATIONS AND OFFICE BUILDINGS
FILTER HOUSE
HOIST HOUSE
STO
SHALE
WASTE SHAFT
SOLUTION CAVITY
SALT
(55 GAL DRUMS)

Source: Battele, Pacific Northwest Laboratories, Reference 2.

FIGURE 9-3

GENERALIZED CONCEPT SOLUTION
MINING FINAL STORAGE FACILITY

formations are examples of the geologic candidates for host rock in this concept. Solidified (borosilicate glass) waste would be received and prepared for disposal. It would then be placed on a combination transporter-hoist vehicle which would move it to the hole and lower it into position. After the hole has received its maximum amount of canisters, it is backfilled and sealed. A generalized concept of such a facility is shown in Figure 9-4. This concept, like most of the concepts described here, features only surface facilities.

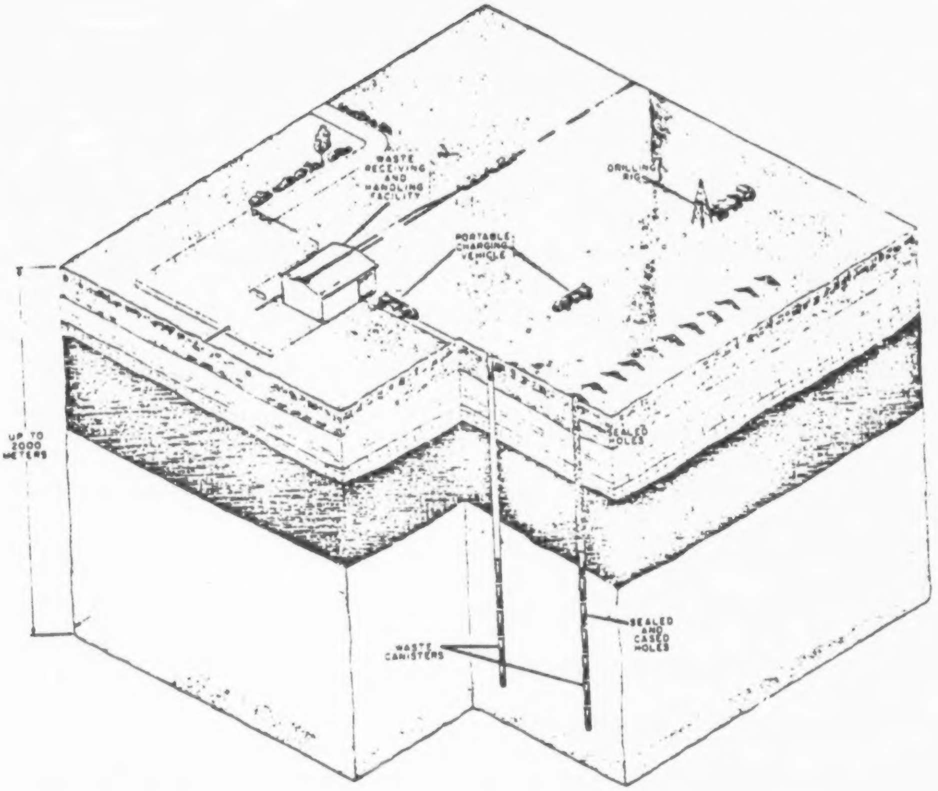
It is assumed that a thick (1000 - 3000m), hydrologically tight, stable formation can be found. The spacing of the holes and of the canister within the holes would have to be designed so that heat can be dissipated without melting.

The problems with this concept lie basically in the many penetrating boreholes which connect the disposal zone with man's environment. It is feared that these boreholes would increase the probability that the integrity of the containment provided by the geological formation could be compromised, with the result that it would be difficult to satisfy the long-term containment requirements.

9.1.3 Waste Disposal in Superdeep Holes

This concept would place waste far from man's environment by placing it in holes which range from 10,000 to 20,000 meters in depth. This great depth would assure that no conceivable climatic or surface change would expose the waste to the biosphere.

The final storage facility using this concept would consist of a large number of large diameter holes drilled into a thick and


WASTE RECEIVING AND HANDLING FACILITY
DRILLING RIG
PORTABLE CHARGING VEHICLE
UP TO 2000 METERS
SEALED HOLES
SEALED AND CASED HOLES
WASTE CANISTERS

Source: Battele, Pacific Northwest Laboratories, Reference 2.

FIGURE 9-4

SOLID WASTE EMPLACEMENT IN A MATRIX OF DRILLED HOLES hydrologically tight sequence of rocks. The waste would be lowered in canisters into the hole until they fill the bottom one or several thousand meters of the hole. After filling the hole to a predetermined level, the hole would be sealed.

The concept has many inherent problems. The technology to drill a large diameter hole to such a great depth, as required by this concept, does not exist today. The time involved in drilling such holes would be close to six years per hole. It is obvious that an enormous financial investment would be necessary to drill the number of required holes and neither the time nor the cost to develop such techniques are known. Another consideration is a limitation on the number of canisters which can be placed per hole if melting of waste and rock is not permitted. Temperature problems are greater as you drill deeper. The rocks may be at a temperature just below melting and the added heat from the waste may induce melting.

9.1.4 Deep-Well Injection

Industry uses deep-well injection for disposal of liquid wastes today. The concept is simple: the liquid waste is pumped down the hole and forced into the geologic formation. Pressures required for pumping range from zero to 10^6 kg/m^2. The host formation must have a porosity of 10-30 percent, a permeability of at least 25 millidarcies*, and a depth of at least 1000 m. The formation must be bounded by impermeable strata and must be free of water-transmitting faults. Such formations occur in the sedimentary basins of the U.S.; however, it is in these basins that oil and gas companies are exploring for petroleum and natural gas. This exploration can cause a major safety problem of connecting waste disposal zones with aquifers. Other important safety factors are proper casing of the injection well and monitoring and maintenance of integrity of all pipes and casings.

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*1 darcy = the passage of 1 cc-per second of a fluid with 1 centipoise viscosity under a pressure difference of 1 atmosphere through a porous medium with a cross-sectional area 1 sq cm and length 1 cm.

Technology needed for this concept is available today; however, its potential for use with liquids containing long-lived or high levels of radioactivity has not been evaluated.

9.1.5 Hydrofracture

Hydrofracture is a concept which is currently being used by industry to either stimulate oil and gas production or for the disposal of wastes. The technology is therefore commercially available.

The concept has three basic steps for the emplacement of waste in a rock sequence such as shale:

Step 1. Breakdown of the geological formation. A viscous fluid which has a gelling and propping agent added to it is pumped under pressure into the well until the formation fractures.

Step 2. Preparation for waste injection. A fluid with a gel breaking agent is pumped in and then drained out, leaving the propping agent behind to keep the fractures open.

Step 3. Waste injection. The waste fluids mixed with a grouting agent are injected into the fractures. The grout hardens and fixes the waste in the formation.

Oak Ridge National Laboratory (ORNL) has used this method of disposal for intermediate level wastes since 1959. The concept has not, however, been demonstrated for high-level, long-lived wastes. As with the prior concept of deep well injection, the long-term containment capability is in question.

9.1.6 Rock-Melting Concepts

The following concepts involve melting of the waste and the surrounding rock. In three of the four concepts, the melted waste and rock are permitted to mix and resolidify as a rock-waste matrix. In the fourth concept, the capsule containing the waste remains intact and melts its way down through the earth's crust. The depth to which the waste penetrates is a function of its aging. Values between 4 and 10 km have been quoted depending on the aging period.

None of the disposal methods involving melting have been extensively investigated, therefore the concepts presented here involving melting are based on preliminary calculations and experiments and, in some cases, conjecture.

9.1.6.1 Mined Cavity/Liquid Waste/Interim Cooling. This concept involves mining a cavity in an isolated, deep (300 to 3000 m) geologic formation (probably an intrusive igneous rock type such as granite) under the fuel reprocessing plant. A cavity having a volume of about 6000 m^3 (a sphere of about 12 m radius) could dispose of 25 years waste from a 5 ton/day reprocessing plant.[3]

After the cavity is formed, waste would be directly injected from the plant. Cooling water would be necessary as the waste begins to boil because of the heat generation of the radionuclides. The steam and other gases created would be collected and sent through a condensation and treatment plant to minimize steam transport of radionuclides. When the cavity is filled with waste, the cooling water will be stopped and all access holes will be sealed. Melting would then follow and would continue for about 65 years, reaching a maximum melt radius of 96 m.

The problems involved with this concept stem from both the emplacement of the liquid waste and from the subsequent melt. During emplacement of the waste, it is necessary to control steam transport of radionuclides to prevent leakage of the waste into an aquifer containing mobile water and possibly to have design features to mitigate buildup of silica scale in the steam exhaust line. After sealing the cavity, steam pressures will build and may cause movement along faults or cracks that may be unknown at the time of emplacement, or may cause new ones to form. When melting begins, the surrounding rock may crack or deform from thermal stress. If this cracking occurs, it may create a new pathway to man's environment.

9.1.6.2 Mined Cavity/Solid Waste/Interim Cooling. In this concept, waste is received from the reprocessing plant in solid form. The waste in canisters is placed in a mined cavity in such a manner as to require interim cooling to prevent melting of the canisters, their contents, and the surrounding rock strata. Cooling would be carried out by filling the cavity containing the waste with water. The cooling water would circulate around the canisters and then to the surface where it would be passed through heat exchangers on the surface. With this concept there is the capability to retrieve any or all the canisters at any time before final sealing and subsequent melting.

After the cavity is full, the cooling water circulation would be stopped and the remaining water would boil away. As soon as the water has boiled away, the waste, canister, and surrounding rock would melt. The rock melt would dilute the waste to a low concentration.

9.1.6.3 Deep Drilled Hole/Solid Waste/No Interim Cooling. This concept places solid waste in deep-drilled (several km below the surface) holes. The host rock would probably be an igneous intrusive type. The waste would be placed in the holes in either expendable canisters or with no canister at all. The heat of decay of the waste melts the waste, the canisters, if any, and the surrounding rock. The waste rock melt mixes by natural convection currents and then resolidifies as it loses heat to the surrounding rock as its heat generation capability decreases. The top of the cavity is then sealed with glass which melts at low temperature. After the glass has resolidified, the remainder of the access hole can be filled with concrete or other suitable material.

This concept has promise but further study is needed in order to fully understand the interaction between the waste and the surrounding rock, both melted and unmelted. Also not fully understood are the long-term radionuclide migration and transport in the host rock; the geologic conditions in deep bedrock; the details of heat transfer; and the transport of volatile and gaseous products from the waste.

9.1.6.4 Solid Waste/Capsule/Deep Descent. This concept calls for a capsule of waste to be placed in a drilled hole up to 2 km deep, which may be partially cased. As the capsule is lowered into position it is cooled by a retrievable cooling system. When in place, the cooling system is shut down and retrieved. The decay heat melts the waste but not the capsule. The capsule transfers the heat to the host rock which melts. Because of its greater density, the capsule settles to the bottom of the melt chamber in a continuing process. The melt at the top of the chamber resolidifies, forming a permanent seal. After a suitable time has elapsed the hole can receive another capsule. This permits one hole to be used for several capsules. The host rock can range from salt domes to intrusive igneous rocks for this capsule.

Problems with this concept are in the area of early capsule failure as well as capsule configuration so as to maximize the amount of waste in each capsule. Capsule size, however, is a tradeoff between several factors, including handling convenience, safety during loading and emplacement, borehole diameter, and thermal properties of the waste.

9.2 Siting (Environmental) Considerations

9.2.1 Geologic, Hydrologic, Climatic, and Other Criteria Which May Affect Long Term Confinement

Concepts for dispersing of high-level radioactive waste will be dependent upon many considerations. These considerations must be dealt with in order to assure safe disposal and effective long-term containment of the waste. The areas of primary consideration which affect the pathways of the radionuclide are as follows:

- Thermal properties of the host rock
- Engineering properties of the host rock
- Water content of the rocks and water movement
- Mineral resources potential
- Geothermal resource potential
- Geographic characteristics
- Seismicity and faulting
- Depth of disposal
- Dimensions of the host rock
- Climate of area and possible changes and their effects on erosion rate

The most suitable rock types for the concepts discussed are 1) intrusive igneous rocks (e.g., granite) or crystalline metamorphic rocks (e.g., quartzite) because of their low permeabilities and high mechanical strengths; 2) salt, either in domes or thick beds because of its low permeability and self-healing properties; and 3) tuffs and shales because of their low permeabilities and high ion-exchange capacities. This list does not intend to imply any preference between the rock types listed above. Sedimentary, except salt and shale, and volcanic rock, exclusive of tuffs, are considered generally unsuitable for waste emplacement because of their potential for high permeability.

Waste form is an important consideration, especially for those concepts which are based upon emplacement of liquid wastes. The geologic restrictions for liquid waste must be more stringent for several reasons:

- higher mobility of the waste in its interim liquid form
- interim manmade barriers (a canister) are not present
- the concentration of waste and its heat are generally higher than for initially solidified waste

An important consideration for concepts involving melting of waste and the surrounding rock is whether or not extensive fractures will develop as a result of the expansion of molten rock. Such fracturing may provide potential pathways to adjacent, possibly permeable, saturated zones. There is also some potential for geysering resulting from the buildup of heat after final sealing of the hole.

9.2.1.1 Thermal Properties of the Host Rock

The dissipation of waste-generated heat is important to the disposal of high-level waste. In order to dissipate heat quickly, efficiently, and steadily, the host rock must have a high thermal conductivity. The conductivity is important in order to minimize surface extent of disposal areas and thereby cost. This is apparent in these concepts for waste disposal with no interim cooling and no melting of either waste or the host rock. In such a concept, a high conductivity would allow more waste per unit area and would thereby help minimize land area needed for the disposal site.

The melting point of the host rock may also hold some significance. In a concept where no interaction of the host rock and waste is permitted, a host rock which has a higher melting temperature is desirable. This would serve to minimize interaction of waste and rock in the event of canister failure. The opposite would hold where the concept calls for the formation of a rock-waste matrix. In this case, a host rock with a lower melting point than the waste is desirable in order to promote rapid mixing of rock and waste.

9.2.1.2 Engineering Properties of the Host Rock

This consideration deals with the mechanical strengths of the host rock. It is obvious that the host rock must have sufficient mechanical strength to allow either mined cavities or drilled holes to remain open during waste emplacement. Rock can fail in many ways; however, we are concerned basically with three modes of failure: rock flow, rock bursting, and rock fracturing. Rock flow occurs when the pressure of overlying layers causes rock to deform plastically. This is common in shales and salt. Rock bursts, as the name implies, occur as sudden releases of stress when the stress becomes greater than the rock's mechanical strength. Fracturing may be more common in the hydraulic-waste injection and deep-well injection disposal concepts. The danger with this mode of failure is the creation of vertical fractures in the rock which could lead to a breach of the host rock and also possibly to a break of waterbearing strata.

Rocks with high mechanical strengths are desirable for disposal of high-level wastes. Rocks which have high mechanical strength and still have generally low permeabilities are granites, gabbros, and quartzites.

9.2.1.3 Water Content of the Host Rock

Groundwater movement is the main pathway by which radionuclides are released into man's environment from disposal areas. It is, therefore, very important that the host rock have as little water content as possible. This includes connate water (water that is formed at the same time as the rock) and fluid inclusions (water trapped during crystallization of minerals).

Site selection must evaluate the possibility of over- and/or underlying aquifers in the vicinity of the host rock under consideration. Where such a situation cannot be avoided, all drilled holes and shafts which penetrate aquifers must be cased and sealed off to prevent movement of material either into or out of the aquifers.

9.2.1.4 Mineral Resource Potential

Exploration for minerals and their subsequent production by future generations can be a potential threat to the long-term confinement of the high-level waste. Site selection for the disposal of these wastes should take into consideration not only the candidate host rock but also rock strata both above and below the host rock. Mineral content and future economic value of the minerals should be determined.

Past mining and/or drilling operations can also jeopardize long-term containment. When the site has been chosen, all past mining and drilling operations must be located so that all mines, shafts, and bore holes can be inspected and properly sealed.

9.2.1.5 Geothermal Resource Potential

With the current search for new energy sources, geothermal energy is being sought and brought on line to help meet electricity and process heat needs. Geothermal energy exploration and development, as with mineral exploration and development, poses a threat to long-term confinement of the waste.

The areas which are thought to be good prospects for geothermal energy are areas which typically have had recent ($< 1 \times 10^6$ yrs) volcanic activity and/or tectonic stresses. For this reason, these areas are undesirable for waste disposal. Also, areas which have above average geothermal gradients are also undesirable because of future geothermal resource potential.

9.2.1.6 Seismicity and Faulting

Seismic and tectonic stability of the rocks in the disposal site is of paramount importance. As has been stated before, all avenues whereby groundwater can penetrate and remove waste must be avoided or sealed off. Crustal cracking and faulting poses a real and great threat to the long-term confinement of high-level waste. It does so by having the potential to rupture the disposal zone and the canisters. In doing so, it can provide excellent pathways for chemical and groundwater removal of the waste and possible exposure to man's environment.

All areas subject to high seismic risk should be eliminated from consideration as possible disposal sites. Sites of lower seismic risk should undergo extensive monitoring and detailed mapping to establish the degree of risk to long-term containment of high level radioactive waste. Only those sites which have the lowest risk should receive further consideration.

9.2.1.7 Depth of Disposal

In general, for a given disposal concept with increasing depth there is greater assurance of long-term containment. There is, however, a need to set a minimum depth at which high-level waste can be disposed of. A minimum depth of 300 meters has been proposed.[4] In areas where this depth would conflict with local water supply aquifers, a greater minimum depth would be required. This would also apply to areas where excessive erosion may occur. These minima are to assure isolation and long-term containment of the waste from man's environment.

As stated earlier, in general, the greater the depth, the greater the assurance of isolation. There are limitations, however. Mined cavities can only be mined to depths where the temperature is low enough to allow man to work. In a typical mine with a geothermal gradient of 20°C/km (20°C at surface), a temperature of 60°C (140°F) is reached at a depth of 2000 meters without artificial cooling. For depths greater than 2000-3000 meters, methods must be used which do not require human entry. The limitations which affect using these greater depths include the temperature at these depths and its effect on canister stability and waste-rock interaction. Also, the degree of difficulty and cost of drilling increase with increasing depth.

9.2.1.8 Dimensions of Host Rock

The dimensions of the host rock should be such that long-term containment can be obtained. In order to do this, the host rock must not only have relatively great thickness but great enough lateral or horizontal extent. Site selection will have to set up minima for these dimensions. Within this specific site selection, the following factors will come into play:

- Total size and shape of host rock formation
- Thickness and extent of surrounding formations
- Homogeneity and isotropy of the host rock
- Thermal properties of host rock
- Hydrological characteristics of both the host rock and surrounding formations
- Waste form
- Chemical properties of host rock and surrounding formations

9.2.1.9 Climate and Possible Changes in Climate

This consideration goes hand in hand with several of the preceding considerations. A dry climate is desirable because it will reduce the amount of groundwater available to leach waste and also reduce the rate of erosion. If such an area is chosen, and there is a change in the climate such that this relatively arid climate becomes a wet rain forest type of climate, the hydrologic regime of the area will change and may pose a threat to the long-term confinement by groundwater leaching and by increasing the rate of erosion.

On a world-wide scale, if world climate becomes warm enough to melt the polar ice caps, either partially or totally, a change in sea level would endanger waste which is disposed of in areas which may be inundated. The opposite is also true. If a new age of glaciation began, any waste buried in areas which may become eroded by glacier movement would have its long-term confinement jeopardized.

A careful analysis of the proposed site must be performed in order to minimize risks to the long-term confinement of the high level radioactive wastes.

9.2.2 Pathways and Barriers of Migration of Nuclides

There are several methods by which the radionuclides can be released from containment and eventually enter the biosphere:

- groundwater intrusion
- faulting
- diapirism
- erosion
- fall of meteorites
- magma intrusion
- change in base drainage levels

These methods of release are minimized before any barriers such as containment vessel and waste form are considered by careful site selection prior to waste emplacement.

There are barriers which can be further used to assure that the radionuclides in the waste do not reenter man's environment during the time required for them to naturally decay to innocuous levels:

- waste form
- canister containment
- geologic system of host rock

In some cases, the barriers must be able to contain the waste for many thousands of years. Such a case is I-129 (half-life 17×10^6 years). Therefore, the probable effectiveness of these barriers will be presented in the following discussion. This will be done as a comparison of barriers and the methods of release and migration.

9.2.2.1 The Waste Form

The waste form will be an important barrier to the migration of radionuclides after canister failure. Various solid waste forms have been considered. These include calcined waste, vitrified (glassification) waste, and waste incorporated in a metal matrix. A borosilicate glass waste form is presently favored both because of its resistance to leaching and the more advanced state of technical development. In the case of borosilicate glass, it has been estimated that "for a cylinder of glass 0.75m high and 0.5 m in diameter, it would take 20 to 200 million years for 99 percent of the initial load of radionuclides to be extracted."[5] For this, it is assumed that the integrity of the cylinder is maintained.

Questions have been raised concerning the long term integrity of the glass form. Heat and radiation range, high pressures, and other factors could result in the failure of the glass form. In the event the glass is fragmented, a greater surface would be exposed and the leaching rate would increase accordingly. In addition, it has been observed that interstitial water migrates towards the heat source in a salt formation. It has also been postulated that the chemical compounds present in a salt formation could form a brine of high leaching capability. It is possible, therefore, that the waste form would provide containment for only tens of years rather than hundreds to thousands of years. Containment would then be dependent upon the host rock.

Groundwater leaching is the chief method of release and migration, and for the long time period involved it is prudent to assume that at sometime groundwater will come into contact with the waste.[7] The other methods of release listed above, as well as accidental access by man, may aid in water contact by providing pathways for water to follow toward the waste. In the event that the waste form can maintain its integrity for hundreds to thousands of years even though the waste is eventually leached out, the time delay will be long enough to eliminate most of the potentially high levels of fission product radionuclides which could find their way back to man's environment.[5,6] The exceptions are the long half-life fission products and activation radionuclides and the actinides. The potential for leaching of radionuclides from a rock-waste melt mix has not been determined.

9.2.2.2 The Canister

The choice of metal for the canister is likely to be from stainless steel, carbon steel, and titanium. Carbon steel and stainless steel are not expected to survive more than a few hundred years, however, it has been suggested that titanium may last for up to 1000 years. It is clear that the canister is not intended to provide containment in the long term. Its role is one of containment in the short term when the high-heat generating fission products are in abundance. The canister also aids in handling the waste during emplacement and recovery, if desired. The largest role may, however, be in preventing rock-waste interaction during the time of highest possible thermal flux which could cause interactions to occur.

The canister will probably be destroyed before about 500 to 1000 years by the geologic environment it is buried in. It is then that the waste form (contained in glass) will become the important barrier. By this time most of the fission products will be gone so that the primary concern is that of migration of long-lived radionuclides. Following loss of the canister and after leaching from the glass, or if the glass is destroyed, the final barrier or delaying action comes into play--the geologic system of the host rock.

9.2.2.3 Geologic System of the Host Rock

The geologic properties of the host rock as stated in Section 8.2 are some of the most important barriers to groundwater leaching of the waste. Site selection must be carried out with three major criteria:

- Hydraulic regime
- Geologic stability
- Retention of radionuclides

The first two of these criteria will eliminate areas which would be highly prone to faulting, diapirism, high erosion rates, magma intrusion, and changes in base drainage levels in the near geologic future. These criteria would also address the permeability of the strata surrounding the host rock as well as the host rock itself. Low permeabilities, along with mechanisms, e.g., ion-exchange capacity, form the host rock's ability for retention of radionuclides. The depth of the waste's emplacement would preclude impact from meteorites as a threat to the repository's integrity.

The ability to retain nuclides by ion exchange is essential to long-term confinement of long-lived nuclides. For the length of time needed to reduce some of the long-lived nuclides to safe levels (e.g., I-129, Np-237, Pu-239), ion-exchange capability can be more important than permeability and depth. Regardless of the host rock's permeability and depth (between the 300 and 6000 m considered here), there is sufficient time for groundwater to penetrate the repository and return the nuclides to man's environment. "Therefore, a geologic formation should not be considered a confining barrier for radionuclides with very long half-lives for which it has no ion-exchange capacity,..."[6]

It has been suggested that it may be possible to artificially set up this ion-exchange capacity in the host rock, adding to its natural capacity, by burying compounds with the waste which would react with soluble ions of the radionuclides to form an insoluble precipitant. This geochemical barrier would provide an additional method of keeping long-lived radionuclides from man's environment for extremely long periods of time. "The greater the ion exchange of the surroundings for a radionuclide, the greater its confinement will be; this confinement may even be total."[6] The physiochemical reactions which will retard the transport of radionuclides include phenomena such as adsorption and colloid filtration as well as ion exchange. The distribution coefficient and retardation factor which are a measure of the sorbtion capability of soils, sediments, and geologic formations were discussed in Section 7 and presented in Table 7-II, for a typical desert soil. Similiar type information is required for specific sites for waste disposal in order to assess their capabilities to provide long-term isolation. Acceptability, however, includes consideration of the initial quantities, the half-life, and the health hazard of the radionuclide as well as the retardation capability of the geological formation. In addition, the various chemical form which the radionuclide may take following leaching from containment and interaction with the host medium must be considered in regard to the sorption effect.

9.3 Technical Feasibility of Alternative Geological Disposal Concepts

The technical feasibility of the concepts helps set up criteria which must be met in order for the concepts to be regarded as viable alternatives to deep mined geologic repositories:

- Achievability with current technology
- Achievability with technology based on current theory
- Ability to provide long-term confinement
- Ability to meet retrievability requirements

It is felt that all of the concepts described earlier can be implemented using extensions of current technology, with the exception of super-deep holes. The technology to drill such deep holes at large diameters does not currently exist. This does not mean that it is not feasible with extensions of current technology. No significant breakthroughs are needed in technology and no uncommon construction, mining, drilling, or operational problems are foreseen with the exception of super-deep holes and with the concepts which call for the formation of a rock waste matrix. Drilling techniques must be developed which will allow drilling of large diameter holes to the depths required for the super-deep concept to become technically feasible. Therefore, all the concepts described seem to be technically feasible using future technology based on current theory and technology. The concepts which involve melting of rock and waste to form a rock-waste matrix need study in the area of the behavior of the molten rock-waste from the time of waste emplacement to the time the rock-waste matrix is solidified in its final disposal form.

Long-term containment is a major concern in the disposal of high-level waste. It is very important, therefore, that all concepts assure long-term containment. The major threat to long-term containment is groundwater. The concepts must preclude contact of the waste with groundwater in order to minimize waste migration to the biosphere. Several of the concepts may have problems with groundwater leaching:

- Disposal in a matrix of drilled holes
- Deep well injection
- Hydrofracture
- Rock melting concepts

The matrix of drilled holes may have a problem because of the many penetrations of the host rock. Each of the drill holes offers a possible pathway to the biosphere. Development and confirmation of sealing techniques would be required.

Deep-well injection, as well as hydrofracture, involves pumping liquid waste into the host rock formation. It is possible that forced injection may form vertical fractures which may give the waste a pathway to waterbearing strata. Techniques of monitoring fracture formation are needed. Although both of these concepts are commercially available, a study of the feasibility of using these concepts for disposal of high-level radioactive waste is needed.

The rock-melting concepts are suspect because of the lack of knowledge of the behavior of the rock-waste melt. Until the uncertainties of its behavior can be resolved, these concepts cannot be considered to assure long-term containment.

Retrievability in high-level waste disposal is very difficult. The concepts involving drilling holes for waste emplacement have limited retrievability as does the solution-mined cavern concept. All of these concepts involve the waste in a solid form at time of emplacement. Hydrofracture and deep-well injection have no retrievability capabilities. Two of the rock-melting concepts have limited retrievability only during interim cooling and emplacement, while the other two have no retrievability. It should be remembered that in final disposal no retrievability is assumed, therefore, this criterion is not of utmost importance.

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